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

October 25, 1990

TO: ALL LICENSEES OF OPERATING NUCLEAR POWER PLANTS AND HOLDERS OF CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS

SUBJECT: GENERIC LETTER 89-10, SUPPLEMENT 3, "CONSIDERATION OF THE RESULTS OF NRC-SPONSORED TESTS OF MOTOR-OPERATED VALVES"

BACKGROUND

In Generic Letter 89-10 (June 28, 1989), "Safety-Related Motor-Operated Valve Testing and Surveillance," the staff of the U.S. Nuclear Regulatory Commission (NRC) requested holders of operating licenses and construction permits to establish a program to provide for the testing, inspection, and maintenance of safety-related motor-operated valves (MOVs) and certain other MOVs in safety-related systems. Supplement 1 to Generic Letter 89-10 (June 13, 1990) provides the results of public workshops held to discuss the generic letter and to answer questions on the staff positions regarding its implementation. In Supplement 2 (August 3, 1990) the NRC staff stated that inspections of program descriptions would not commence until January 1, 1991, and, thus, the program descriptions need not be available on site until that date.

In parallel with the NRC staff's activities leading to Generic Letter 89-10, the staff performed tests of MOVs as part of an ongoing research effort. The tests were conducted on 6-inch and 10-inch gate valves typically used to provide containment isolation in the steam supply lines of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, and in the supply line for the Reactor Water Cleanup (RWCU) system at boiling water reactor (BWR) nuclear power plants. On June 5, 1990, the staff issued Information Notice 90-40, "Results of NRC-Sponsored Testing of Motor-Operated Valves."

As discussed in Information Notice 90-40, the NRC-sponsored tests revealed that, regardless of fluid conditions, the tested valves required more thrust for opening and closing under various differential pressure and flow conditions than would have been predicted from standard industry calculations using typical friction factors. Thus, although the NRC-sponsored tests focused on the HPCI, RCIC and RWCU containment isolation valves at BWR plants, the information obtained from those tests may be applicable to valves used in other systems at BWR and pressurized water reactor (PWR) plants. For example, calculations using low valve friction factors may underestimate thrust requirements for opening and closing valves.

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In response to a staff request, the BWR Owners' Group obtained information from the BWR licensees regarding the capability of MOVs used to provide containment isolation in the steam lines of the HPCI and RCIC systems, and in the supply line of the RWCU system. The staff's review of the NRC-sponsored test results and the MOV data provided by the BWR Owners' Group indicates that deficiencies might exist in those MOVs.

DISCUSSION

In Generic Letter 89-10, the NRC staff requested that licensees and construction permit holders complete the programs established in response to the generic letter (excluding the periodic verification of MOV switch settings) by June 28, 1994, or within 3 refueling outages after December 28, 1989 (or operating license issuance for construction permit holders), whichever is later. While recommending that licensees and permit holders consider the safety significance of MOVs in developing their programs, the staff did not have sufficient information at that time to recommend that licensees and permit holders establish any particular priority for MOVs within the generic letter program. The information recently obtained from the NRC-sponsored tests, however, may affect the priorities being established by licensees and permit holders for implementing their generic letter programs. From its evaluation of the MOV data provided by the BWR Owners' Group and the results of the NRC-sponsored tests, the staff has determined that correction of any deficiencies in the HPCI, RCIC and RWCU MOVs described herein need to be given high priority in the implementation of generic letter programs. While such deficiencies may not need to be corrected immediately, the staff has determined by means of a safety assessment (Enclosure 1) that any MOV deficiencies should be corrected within 18 months or by the end of the first refueling outage, following issuance of this generic letter supplement, whichever is later. The staff's review of a generic safety assessment performed by the BWR Owners' Group (Enclosure 2) confirmed that this time period is acceptable for correcting any deficiencies in those MOVs. If a BWR licensee believes that there are MOVs with potential deficiencies at its facility that have greater safety significance than the HPCI, RCIC, and RWCU MOVs described herein, the licensee should determine the appropriate priority for completing the generic letter program for those valves.

REQUESTED ACTIONS

BWR licensees are requested to assess the applicability of the data from the NRC-sponsored MOV tests, to determine the "as-is" capability of the HPCI, RCIC, and RWCU MOVs described herein, and to identify any deficiencies in those MOVs. Where applicable, BWR licensees should also evaluate the MOVs used for containment isolation in lines to the isolation condensers. Elements that a BWR licensee may consider in determining whether the NRC-sponsored test data are inapplicable to its HPCI, RCIC and RWCU MOVs include valve size, type and manufacturer; disk type; design-basis differential pressure and flow conditions; internal dimensions and clearances; and disk and guide surface materials.

BWR licensees are requested to perform a plant-specific safety assessment to verify that the generic safety assessments performed by the NRC staff and the BWR Owners' Group are applicable. In performing the plant-specific safety assessment, BWR licensees should address factors such as consideration of functional valve test results; operating procedures and emergency operating procedures; the conduct of training; current torque switch bypass settings including the potential for motor overload on a first attempt to close the valve; leak detection capabilities; inspection programs for erosion-corrosion and intergranular stress corrosion cracking (including response to Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping"); water-hammer prevention practices; the environmental qualification of the MOVs and other nearby equipment; radiological consequences both on and off the plant site that could result from a pipe leak or break; and probabilistic risk considerations. Where applicable, BWR licensees should include in their plant-specific safety assessments MOVs used for containment isolation in lines to the isolation condensers. If a BWR licensee believes that there are MOVs with potential deficiencies at its facility that have greater safety significance than the HPCI, RCIC and RWCU MOVs (and the MOVs in the isolation condenser lines) described herein, the licensee should justify as part of its plant-specific safety assessment the prioritization of its effort to identify and correct MOV deficiencies.

BWR licensees should consider the implementation of short-term corrective actions. For example, BWR licensees should evaluate the feasibility of increasing torque switch settings where the motor, actuator, and valve are designed to accommodate such an increase. BWR licensees should develop procedures and provide training for plant personnel to respond to a pipe leak or break in a line containing a deficient MOV, particularly if the deficiency cannot be corrected in the short term.

BWR licensees may accomplish these recommendations as part of an accelerated response to Generic Letter 89-10 for the applicable MOVs. For example, BWR licensees could complete the design-basis reviews for those MOVs and could establish torque switch settings as described in Recommended Actions a and b of the generic letter, respectively. Recommended Action c of the generic letter requests that the MOVs be tested in situ under design-basis differential pressure and flow conditions, where practicable. For those instances where design-basis testing in situ is not practicable and an alternative to such testing cannot be justified at this time, the staff recommends that the BWR licensee use the "two-stage" approach discussed in Generic Letter 89-10 and Supplement 1. Following that approach, the BWR licensee would determine the operating requirements of the MOV using the best data currently available and then obtain applicable data as soon as possible.

While the reporting requirements below are addressed to BWR licensees, all licensees and construction permit holders should consider the applicability of the information obtained from the MOV tests and the staff evaluation of the test results to other MOVs within the scope of Generic Letter 89-10. In addition, all licensees and permit holders should consider this information in the development of priorities for implementing the generic letter program.

REPORTING REQUIREMENTS

In order for the NRC to determine whether any BWR operating licenses should be modified, suspended or revoked, BWR licensees shall provide written information, signed under oath or affirmation pursuant to Section 182 of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), as follows:

1. Within 30 days of the receipt of this letter, BWR licensees shall notify the NRC staff that a plant-specific safety assessment report addressing, as a minimum, the factors described herein, is available on site for staff review. BWR licensees shall also notify the NRC staff whether they believe that there are MOVs with deficiencies of greater safety significance than the MOVs used to provide containment isolation in the steam supply lines of the HPCI and RCIC systems, in the supply line of the RWCU system, and in the line to the isolation condenser.

2. Within 120 days of the receipt of this letter, BWR licensees shall provide to the NRC staff the following:

   a. Criteria, reflecting operating experience and the latest test data, that were applied in determining whether deficiencies exist in the HPCI, RCIC and RWCU MOVs described herein, in the MOVs in isolation condenser lines, and in any MOVs considered to be more safety significant, as applicable;

   b. The identification of any MOVs found to have deficiencies; and

   c. A schedule for any necessary corrective action.

3. Subsequent to the determination of necessary corrective actions or the establishment of the schedule for completion of those actions, BWR licensees shall inform the NRC staff of any changes to the planned actions or schedule.

As noted above, based on the generic safety assessments prepared by the NRC staff and the BWR Owners' Group, the staff believes that justification exists for individual plants to which those safety assessments are applicable to take 18 months or to the end of the first refueling outage, following issuance of this generic letter supplement, whichever is later, to resolve any deficiencies in the HPCI, RCIC and RWCU MOVs described herein. However, if a BWR licensee determines that a more limited time is mandated by its plant-specific safety assessment, the licensee should utilize the more restrictive time. If additional time is needed to complete the corrective actions, BWR licensees should submit the plant-specific safety assessment and obtain staff approval for the corrective action schedule.

BACKFIT DISCUSSION

Based on operating experience and research results, the staff determined several years ago that MOV tests beyond those previously acceptable are necessary to satisfy the NRC regulations. As that determination constituted a backfit, the staff prepared Generic Letter 89-10 in accordance with NRC procedures for the issuance of staff guidance containing backfit provisions. Supplement 3 represents a further backfit in that the staff is requesting BWR licensees to advance the schedule for Generic Letter 89-10 with respect to specific MOVs at BWR plants. This limited advancement of the Generic Letter 89-10 schedule is the result of the information obtained from NRC-sponsored MOV tests indicating that deficiencies might exist in certain MOVs installed to perform containment isolation functions at BWR plants. The staff has determined that the issuance of Supplement 3 to Generic Letter 89-10 is necessary to provide confidence that BWR facilities are in compliance with their safety analyses and NRC regulations such as described in 10 CFR Part 50, Appendix A, Criteria 54 and 55. More specifically, because deficiencies might exist in the MOVs described herein, the staff does not have adequate confidence that (1) as required by Criterion 54, the applicable piping systems which penetrate containment have been provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems, or have been designed with the capability to test periodically the operability of the isolation valves and associated apparatus or (2) as required by Criterion 55, appropriate requirements, such as higher quality in design, fabrication, and testing, to minimize the probability or consequences of an accidental rupture of lines which are part of the reactor coolant pressure boundary and penetrate reactor containment have been provided as necessary to assure adequate safety. Therefore, the staff has determined that the backfit provisions of this generic letter supplement are justified under 10 CFR 50.109 (a)(4)(i). Based on its safety assessment, the staff determined that no immediate corrective actions are needed and that BWR licensees may proceed to resolve any deficiencies in the MOVs described herein as recommended in this letter.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires December 31, 1991. The estimated average burden hours are 150 person-hours per licensee response, including assessment of the new recommendations, searching data sources, gathering and analyzing the data, performing data evaluations, and preparing the required letters. (These estimated average burden hours pertain only to the identified response-related matters and do not include the time for actual implementation of the requested action.) Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, Division of Information Support Services, Office of Information Management, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

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Enclosures: As stated

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Enclosure 1

NRC STAFF SAFETY ASSESSMENT RELATIVE TO BWR CONTAINMENT ISOLATION VALVES FOR HPCI, RCIC, AND RWCU

On July 6, 1990, the BWR Owners' Group supplied the NRC staff with the results of a survey of MOV data on the containment isolation MOVs in the steam supply lines of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems and in the supply line for the Reactor Water Cleanup (RWCU) system. An evaluation of the MOV data indicated that about a third of these valves may not be able to isolate the blowdown flow from a postulated pipe break. This document was prepared to evaluate the safety significance of these potential deficiencies.

LIKELIHOOD OF PIPE BREAK

HPCI and RCIC Low Erosion/Corrosion Susceptibility

The HPCI and RCIC steam lines are fabricated from ferritic steel. Due to the erosion/corrosion susceptibility of ferritic steel in the BWR environment, NRC issued Generic Letter 89-08, "Erosion/Corrision-Induced Pipe Wall Thinning." This generic letter recommends that all licensees establish an erosion/corrosion monitoring program to assure the structural integrity of high-energy carbon steel piping systems. To establish pipe inspection frequencies, licensees have performed various predictions of erosion/corrosion susceptibility. The HPCI and RCIC steam lines are used only intermittently during pump testing. For this reason, the predictions indicate insignificant erosion/corrosion will occur in these lines. Most licensees contacted, during a brief survey on this subject, are not planning inspections of these lines.

RWCU Augmented Inspections

The RWCU supply lines are fabricated from austenitic stainless steel. Since austenitic stainless steel in the BWR environment is susceptible to intergranular stress corrosion cracking (IGSCC), NRC issued Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." Accordingly, licensees have implemented programs to perform augmented inspections of BWR piping fabricated from austenitic stainless steel. However, it is not evident at this time how many licensees have implemented augmented inspection programs for RWCU lines outside containment. Licensees have received plant specific requests for information (RAI) regarding their compliance with the inspection requirements of GL 88-01 and will be expected to address the inspection of these lines in response to the RAI. Furthermore, licensees are expected to address the status of their inspection of the RWCU lines in the plant specific safety assessment.

Piping Stress Levels

All of the safety-related piping in the systems under discussion have been designed to applicable ASME Section III rules or similar ANSI B31 rules. Implicit in the allowable stresses of these Codes is a substantial built-in margin below the material ultimate strength. Furthermore, the normal operating stresses in the systems under discussion are generally much lower than the stresses allowed by design, which further decreases the probability of failure in this piping.

Failure Mechanisms

Nuclear and fossil power plant experience has indicated that large breaks have resulted from either large water hammer events or undetected significant pipe wall erosion. We believe that there is a low probability of these mechanisms occurring in the subject piping. The augmented inspections being performed for detection of erosion are discussed above. The technical findings relevant to the resolution of Unresolved Safety Issue A-1, Water Hammer, were contained in NUREG-0927, Revision 1, "An Evaluation of Water Hammer Occurrence in Nuclear Power Plants." In this NUREG the safety significance of water hammer in the RCIC and RWCU systems is classified as low. Taking into account changes that have been made in plant operating procedures, because of the results reported in this NUREG, the safety significance of water hammer in the HPCI system is now also considered to be low. Therefore, the probability of a large pipe break in any of the subject lines should be low. Furthermore, should a leak develop, it is likely to be detected by quadrant temperature and floor drain sump level monitors. These monitors alarm in the control room and cause entry into annunciator response procedures. These procedures would direct the operators to determine the cause of the alarm and would lead to closure of the MOV in the leaking pipe before a break could occur.

PLANT MITIGATIVE FEATURES

Margin on Assumed Differential Pressure

The differential pressure assumed in the design phase for the establishment of the operating capability of the MOV might be greater than would actually occur during a blowdown event. This may provide some additional thrust for valve closure during a blowdown from a line break.

Valve Redundancy

The HPCI and RCIC steam supply lines and the RWCU letdown lines are all equipped with two motor-operated isolation valves, one inside containment and one outside containment, with one powered by an AC motor and the other by a DC motor. These pairs of valves receive coincident signals to close and have the same static load closing stroke times.

DC motors may slow down considerably under high loadings resulting in a longer stroke time. Nevertheless assuming that both valves have power to close, the total differential pressure between the reactor and the break would be shared across the two valves. Due to the possibility that these valves are not set up with adequate thrust for a single valve to close against the differential pressure of a blowdown, the reduced pressure load could increase the likelihood of one or both valves closing.

Closure After Depressurization

For a large HPCI or RCIC steamline break, if the isolation valves fail to close, the reactor will depressurize below the low pressure injection systems shutoff head before the core would begin to uncover. With offsite power still available, the condensate pumps can likewise provide abundant cooling at low pressures to maintain core integrity. After depressurization the load on the isolation valves would be greatly reduced. For MOVs that failed to close completely because of a trip of the torque switch or a thermal overload device, the likelihood of closure on the second try would be high.

For the case of an RWCU system break, if the isolation valves fail to close, HPCI and RCIC would both be available to provide some make up while the reactor coolant system depressurizes. However, these systems alone would not keep up with the loss from larger RWCU breaks. Depressurization through the break or by ADS would result in core uncovery until low pressure systems refill the vessel. With the low pressure systems functioning, however, significant core damage should not occur. As discussed above, after depressurization the load on the isolation valves would be greatly reduced and the likelihood of closure of an undamaged MOV on the second try would be high.

The likelihood of valve reclosure in any of these lines would be low if the valve motors are undersized. In addition, the above discussion assumes a degree of separation of emergency core cooling systems that may not be present for all BWR designs. Licensees should address these aspects in a plant-specific safety assessment.

Consequence Mitigation

The primary symptom in the emergency procedures for a break in one of the lines under consideration is water level. The mitigative systems for supplying make up are HPCI and/or RCIC, main feedwater, low pressure core injection and other low pressure systems that can be used in an accident management capacity. Provided adequate make up water is available, core cooling would continue without serious offsite consequences even if isolation of the broken line is delayed until much later in the scenario.

RISK PROBABILITY ANALYSIS

The initial view of staff risk analysis experts is that the identified MOV concerns should be resolved promptly, but that taking immediate action resulting in BWR plant shutdowns is not justified. The NRC staff is performing a sensitivity analysis using the NUREG-1150 models based on the identified deficiencies to gain insights on this problem and will determine the need for more detailed modeling. Preliminary results of the sensitivity analysis are expected to be available by the end of October.

CONCLUSIONS

Given that some of the valves in the subject lines presently may not be capable of closing under the design basis differential pressure, we believe that the factors discussed above could be used to justify continued plant operation while licensees expeditiously pursue corrective actions. It is recognized that some corrective actions may take as long as a refueling cycle to implement.

ENCLOSURE 2

SAFETY ASSESSMENT

ISOLATION FUNCTION OF MOVs
FOR HPCI AND RCIC STEAM SUPPLY LINE
AND RWCU WATER SUPPLY LINE

GE NUCLEAR ENERGY
FOR
BWR OWNER'S GROUP

JULY 1990
AUGUST 1990 (REVISION 1)

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

The only undertakings of General Electric Company (GE) respecting information in this document are contained in the applicable contracts between GE and the BWR Owner's Group utilities as specified in GE Proposal 355-1951, Rev. 3, accepted by the respective participating utilities' Standing Purchase Order for the performance of the work described herein, and nothing contained in this document shall be construed as changing those individual contracts. The use of this information except as defined by said contracts, or for any purpose other than that for which it is intended, is not authorized; and with respect to any such unauthorized use, neither GE or any of the contributors to this document makes any representation or warranty, and assumes no liability as to the completeness, accuracy or usefulness of the information contained in this document.

1.0 Introduction

On June 7, 1990 the NRC, by letter to the BWR Owners' Group (BWROG), requested data concerning certain safety-related BWR Motor Operated Valves (MOVs) capabilities. Data was requested for the primary containment isolation valves in the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) steam supply lines, and the Reactor Water Clean-Up (RWCU) suction lines. This request was the result of a BWROG and NRC May 24, 1990 meeting. This meeting concerned the applicability of the Idaho National Engineering Laboratory (INEL) test data performed to resolve Generic Issue 87. The NRC interpretation of this data is in Information Notice 90-40 "Results of NRC-Sponsored Testing of Motor-Operated Valves" dated June 5, 1990.

The NRC interpretation of the test results appeared to indicate a 0.3 disk factor, normally used to calculate valve seating forces, is not conservative. The calculated valve seating force is used to size the valve actuator and motor, and set the torque switch. Therefore, the actuator size or torque switch setting may be marginal or may not fully close the valve against postulated maximum design basis event flow and differential pressure (dp). This safety significance assessment, requested by the BWROG, documents the adequate safety margin of BWR plants. It shows a significant safety concern does not exist, even if the HPCI, RCIC and RWCU isolation MOVs of concern may not have optimally sized or set actuators for full closure under postulated maximum design basis event flow and dp conditions.

2.0 Summary

The isolation MOVs of concern were selected, sized, and set using good engineering judgement based on the state of the art at the time of purchase. On a plant specific basis, features were provided for early means of leak detection before a complete design basis pipe failure could occur. In addition, other systems which provide additional valve isolation capability are available. Materials were selected for low probability of pipe failure. In Service Testing in conformance with plant Technical Specifications is performed on the piping and valves to confirm their suitability and readiness for service. Four of the six

subject valves have been evaluated and tested based on IE Bulletin 85-03 [9]. Emergency Procedures Guidelines for other diverse plant systems provide means of rapidly reducing the MOV service conditions if a pipe break occurs.

It is recognized that INEL testing has identified anomalous valve behavior in the test valves under their test conditions. The BWROG and utilities are following this testing and reviewing engineering data as it becomes available for plant application. Based on the data applicability to their plant and equipment capabilities, utility personnel are reviewing their MOVs to assure the valves will operate on demand under all possible conditions.

This assessment employing a realistic integrated systems approach concludes existing BWR MOVs for HPCI, RCIC and RWCU systems supply line or suction line isolation have a very high probability of full isolation under realistic conditions. In addition, HPCI and RCIC steam and the RWCU water supply line MOVs have demonstrated proper operation under conditions mimicing the likely demand event, a pipe leak. System isolation will occur before the postulated design basis event high flow dp condition. Based on this the presently installed and set equipment does not represent an undue risk to the health and safety of the public.

In process utility actions responding to GL 89-10 are proceeding with consideration of the INEL data to prioritize valves for review and testing.

Individual plant licensing documents (SARs) have established that pipe cracks produce leaks long before pipe failure would be expected. In addition, the NRC has accepted this conclusion when approving the leak-before-break concept as a basis for pipe restraint removal in Light Water Reactors.

Leak detection equipment exists at all BWR plants to detect the small pipe leak condition and then to initiate system isolation. Small leaks represent such a small quantity of fluid flow escaping from system piping that normal system flow parameters will not be noticeably changed. The system flow conditions during a small leak will remain almost the same as the system normal standby and operational conditions.

These environmentally qualified MOVs, which perform the isolation function, have shown adequate operability for many years during normal, periodic, operational testing and inadvertent isolations. The most probable, realistic, safety (isolation) response required of these MOVs will be from a postulated pipe leak condition outside the containment. The likelihood of a leak occurring in these lines is small. Even if a leak occurred it would be detected well before a high flow/dp condition develops. Substantial time exists for detection of such a pipe leak and completion of the isolation function by valve closure.

The MOV isolation performance will be the same as already demonstrated by multiple isolations (both during periodic testing and inadvertent initiations) of these valves in most operating plants.

A realistic assessment of the consequences of a postulated design basis pipe break condition, or some intermediate pipe break condition, leads to the conclusion that there is adequate safety margin to protect the reactor core and isolate the system successfully. Any single ECCS pump is adequate to provide core cooling. Analysis has shown any single low-pressure pump (i.e., RHR or core spray) has adequate capacity to overcome the inventory loss associated with the postulated failure in one of the lines in question. Additionally, the HPCI, RCIC and RWCU lines are equipped with two isolation valves. If either of these closes isolation is accomplished. Any action which reduces the differential pressure across either valve will allow system isolation. Some of these actions include partial valve closure, depressurization through the postulated break and/or primary system depressurization as directed by the Emergency Procedure Guidelines (EPGs).

It is not expected HPCI/RCIC/RWCU system isolation MOVs will be challenged at high flow design basis accident conditions because of leak-before-break considerations. Leaks should be isolated early at low flow conditions due to the effective leak detection and isolation systems. There is a significant high probability of successful valve closure when realistic consideration of expected plant and system responses to postulated accident conditions are used. Reactor coolant inventory losses can be made up even without successful full valve closure for a postulated rupture in these lines. There is adequate safety margin in the ECCS to handle the losses. The ECCS are designed for a much larger break than these small line ruptures. 10CFR100 off site dose limits are not expected to be exceeded even with a delayed isolation response for any of these three systems.

3.0 Safety Assessment - HPCI/RCIC/RWCU Pipe Leaks

3.1 Leakage Considerations

It is industry experience that high energy pipes experience leaks long before a pipe break condition develops. Industry has referred to this phenomena as Leak-Before-Break (LBB). Most BWR plants have multiple channel, redundant leak detection monitoring of the high energy system lines external to the containment. This monitoring is sensitive to small leaks and causes both an alarm in the control room and at most plants automatic isolation signals to the leaking system's isolation MOVs. Isolation signals or operator action would initiate MOV closure long before the leakage could cause any significant flow change, fluid loss or radiation release, and before a significant long term environmental challenge to the MOVs. The MOVs have been environmentally qualified to the more extreme Double Ended Guillotine Break (DEGB) environmental conditions. The MOVs are periodically inspected and tested to demonstrate operability during plant operation. In addition, these valves have occasionally been inadvertently closed during plant operation. This has demonstrated unscheduled demand operability.

3.2 Leak-Before-Break Justification

Although the design basis for nuclear power plants, as discussed in the SAR, includes the evaluation of a loss of coolant accident resulting from a postulated pipe break, considerable effort goes into designing piping and safe end systems to assure that such a break will not occur. Piping systems are analyzed using appropriate codes and standards, typically Section III of the ASME Code, to limit applied stresses, and materials are selected to provide adequate ductility and toughness. Piping design also provides implicit margins concerning fatigue initiation. Environmental effects are not considered significant. Piping materials (carbons steel in most cases) and steady state temperatures (less than 250°F in many cases) preclude environmentally assisted cracking. Thus, while cracking may be postulated, the probability is low. Furthermore, leak detection systems are designed to assure that, even if a pipe or safe-end (nozzle-pipe transition piece) should experience cracking, the crack would grow to a through-wall leak and the leak would be detected well before it reaches critical crack size which could cause a pipe rupture in the long term. This concept is called the 'Leak-Before-Break' concept or approach. This critical crack basis already exists in most plant SARs as part of the plant design basis discussion. In more recent plants it is typically covered in Chapter 5 of the SAR.

In general terms, the LBB concept is based on the fact that reactor piping and safe ends are fabricated from tough ductile materials which can tolerate large through-wall cracks without complete fracture under service loadings. By monitoring the leak rate from the through-wall cracks and setting conservative limits on the leakage, cracks in piping can be detected well before the margin to rupture is challenged.

In NUREG 1061, Volume 3 [1], the NRC Piping Review Committee outlined the limitations and general technical guidance on LBB analyses to justify mechanistically that breaks in high energy fluid system piping need not be postulated. In a recent modification to General Design Criterion 4 [2], the NRC has formalized the use of the LBB approach to justify the elimination of pipe whip restraints and jet impingement barriers as design requirements for a hypothetical DEGB in high energy reactor piping systems. Thus there is NRC recognition the LBB concept provides added margin over and above the ASME Code piping design structural margins.

A key parameter in the LBB evaluation is the critical crack length at which pipe rupture is predicted. The focus in the LBB evaluation is on the through-wall circumferential cracks because such cracks could lead to a DEGB. A DEGB is one of the usual design basis event analysis assumptions.

The LBB approach is not being applied in this assessment to eliminate pipe whip restraints or jet impingement barriers or reduce inspections. Therefore, explicit LBB margins are not calculated nor are they necessary. Instead, the LBB concept is used in this assessment to demonstrate that the leakage from a through-wall crack with a length up to but less than the critical crack length, would be large enough to be readily detected such that isolation actions can be taken well before the critical crack length is achieved and long before maximum design basis event flows and pressures are established.

3.3 Critical Crack Length and Leak Rate Calculations

Critical crack length and leak rate calculations for typical BWR piping geometries have been documented in plant SARs. Reference 3 is an example of such calculations. The calculations presented here use methods [4,5,6] more recent than used in the existing SAR calculations.

Table 1 lists the values of parameters used in the critical crack length and leak rate calculations. The results of the calculations for representative pipe sizes are summarized in Table 2. A limit load approach with a conservative value of flow stress equal to 2.4 S_m (where S_m is the value of material design stress intensity given in the ASME Code), was used in calculating the critical crack lengths. When based on test data, the flow stress for four inch diameter pipes was assumed to be 2.7 S_m. The leak rate calculation methods used for both the water and the steam lines are outlined in Reference 5.

An inspection of Table 2 shows that the calculated leak rate at critical crack length is, as expected, a strong function of pipe diameter. Nevertheless, even for the 4-inch diameter water line, the predicted leak rate is 25 gpm at close to the critical crack length. A 25 gpm leak rate is larger than the leak detection rate sensitivity identified in the following section on Leak Detection with the exception of the RWCU cold water lines. These calculations conservatively ignore leak rate increases due to steam cutting, that can occur for a given crack length. Once leakage starts, due to steam cutting, it increases with time and the Table 2 leak rates can occur before reaching critical crack length. Full design basis MOV dp, corresponding to a DEGB, will not occur at these limits due to the down stream flow restriction (crack). Thus complete MOV closure will occur under these conditions. The RWCU cold lines have a much lower potential for cracking because of their constant cold condition and materials.

It is important to emphasize that the LBB margin increases with increasing pipe size. Thus, larger pipes where failure could be significant have inherent LBB advantages. While the LBB margin is somewhat lower for smaller pipes, there is still a large BWR experience database supporting the integrity of such piping.

Inspection programs (e.g., In Service Inspections (ISI) per ASME Section XI), other Generic Letter 88-01 [8] commitments and other periodic inspections on system piping outside the isolation valves provide additional assurance of continuing piping integrity and low probability of pipe leak and break conditions.

Based on the results of this and the following evaluation, it is concluded that the subject piping systems (HPCI, RCIC Steam Supply Line and RWCU Water Supply Line) are expected to develop a detectable leak long before reaching the point of incipient rupture. Thus, a DEGB in these lines is highly unlikely.

3.4 Leak Detection Monitoring and Isolation

Most BWRs have been designed for compliance to General Design Criterion (GDC) 54 [7] - "*Piping system penetrating containment.* Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities ...." This GDC was satisfied with a defense in depth combination of pipe break, high flow monitoring and isolation sensors for large leaks for each high energy piping system. These same high energy piping systems also have sensitive, small leak, temperature monitoring and isolation sensors.

At most plants the redundant, safety grade temperature monitoring equipment continuously monitors areas outside containment where high energy lines are routed. The temperature sensors for this monitoring are grouped with the piping of each system and will alarm and/or isolate that system when a leak condition is detected. At most plants the sensors and logic are applied in a redundant design configuration to be single failure tolerant. These temperature sensors can be configured in an ambient temperature and a differential temperature arrangement. The configuration is room dependent at each plant.

The range of plant system area construction differences has resulted in alarm and isolation limits related to leaks typically from 5 gpm to less than 25 gpm. These isolation limits are converted to temperature values, and are expressed in terms of temperature in SAR Technical Specifications and other plant documentation. The temperature sensors sensitivity provides a fast response to a developing leak. Even though a temperature limit may relate to a specific leak rate, these same temperature limits can be attained with much lower leak rates. A smaller leak for a longer time period can reach the temperature limit too and allows recognition of smaller cracks.

In addition to temperature monitoring in the RWCU system, most plants have cold water low flow leakage monitoring capability. This cold water, small break, redundant, safety grade, differential flow monitoring leak detection capability measures flow in to and out of the system. It has an isolation limit of less than 100 gpm flow mismatch between the system input and its outputs. It can quickly respond to a small break condition in the cold water portions of RWCU. Typically this isolation limit would initiate MOV closure before any appreciable additional flow could be developed. The RWCU heat exchangers dp drop will further limit any small break flow. This monitoring sensitivity has been inadvertently demonstrated numerous times during start-up and realignment of the RWCU system.

In addition to the temperature monitoring system and the differential flow monitoring (RWCU), the operator can detect small leakage flow into the area or equipment room drain Radwaste sumps. There are also area radiation monitoring system gamma detectors that may alarm during small leak conditions. These additional leakage information sources provide data to the operator which call for a visual inspection of the area.

Operating experience has shown relatively quick operator response to leaking conditions in safety systems and other monitored systems upon leak identification by routine inspection activities or by monitoring equipment isolations and alarms.

The leak detection temperature monitoring capability installed in BWRs can detect the small leakage condition and initiate isolation long before a pipe break condition would develop. Therefore, the combination of the leak-before-break approach in conjunction with the leak detection capability provides early isolation at less than design basis conditions for a potential pipe break that might challenge the MOVs isolation capability at maximum flow induced dp.

3.5 Radiological Consequences of Leakage Flow

The radiological consequences of the leakage flow from the HPCI, RCIC or RWCU lines are bounded by the plant design basis radiological release. The BWR design basis event for offsite release is the DEGB of the main steam line. The DEGB assumed in the evaluation of the offsite release results in a large amount of reactor inventory loss prior to break isolation. The liquid phase of the reactor inventory contains most of the radioactive material which is released into the secondary containment during the postulated break event. However, the resulting dose from the main steam line break is still only a fraction of the 10CFR100 limits. Furthermore, the total inventory loss for the small leakage associated with the HPCI, RCIC or RWCU line is only a small fraction of that from a main steam line DEGB. For example, a 25 gpm hot water leak from RWCU typically can be detected within 10 seconds. This means that the total inventory release before detection is less than 30 lbs. This is a small fraction compared to the main steam line break liquid inventory loss which is approximately 140,000 lbs total, of which 120,000 lbs is liquid. Therefore, even if the leak detection requires 4000 times longer to isolate the detected leak, the radiological release from the leakage flow will be a very small fraction of the 10CFR100 limit.

3.6 Environmental Qualification

Equipment Qualification (EQ) is evaluated on a plant specific basis. The EQ programs have established the capability of the plant safety related electrical equipment to perform their design basis safety functions under the limiting environmental conditions postulated for that equipment. Typically, equipment is qualified based on type testing at bounding environmental conditions that envelope a broad range of applications. MOVs are typically qualified to environmental envelopes that bound HELB conditions as well as in containment LOCA conditions. Other required equipment will also be qualified to at least the limiting HELB conditions which are much worse than the small leak environmental conditions that would be postulated due to the leak before break scenerio.

Assuming the design basis DEGB were to occur with delayed isolation the resulting environmental conditions affecting required equipment would still be expected to be bounded by existing EQ evaluations. Since essentially atmospheric pressure would exist in the reactor building/auxiliary building the maximum achievable temperature in the vicinity of the postulated break (i.e., 212°F) would be the same with or without prompt isolation. Time duration at maximum conditions would increase without prompt isolation but is still expected to be bounded by existing test envelopes. The majority of the required safety related equipment (basically ECCS equipment) would typically have been qualified to envelopes that bound in containment LOCA applications, this would include equipment such as MOVs, cabling, and transmitters. ECCS motors are typically qualified for saturated steam conditions at temperatures up to 250°F.

The most limiting equipment is expected to be switchgear and motor control centers. For highly compartmentalized building arrangements typical of Mark II and Mark III containment designs these equipment will be located in areas not appreciably affected by the pipe breaks being postulated. For noncompartmentalized arrangements the bulk building conditions could be postulated to reach saturated steam conditions at atmospheric pressure if the pipe break is not isolated. These conditions could exceed existing qualification limits for such equipment and should be evaluated on a plant specific basis. One typical plant has been examined and it was determined that the existing qualification envelope of this equipment remained bounding. The final potential area of concern that must be considered is equipment submergence due to flooding resulting from a postualted pipe break without isolation For building arrangements that would be subject to such flooding it is expected that existing flood control measures such as curbs and drains are adequate to control the volume of water released from such breaks. This should be confirmed on a plant specific basis.

Therefore, no EQ concern exists for MOV isolation or the functioning of other safety systems equipment due to small pipe leaks postulated under the leak before break criteria. In evaluating the consequences of a postulated DEGB without prompt isolation it is expected that existing EQ envelope conditions will remain bounding, subject to plant specific confirmation of submergence assumptions and switchgear/MCC qualification in noncompartmentalized building arrangements.

3.7 Leakage Flow and Inadvertent Closure

From leak-before-break considerations and with the capabilities of detection and isolation of a small leak, the leakage flow from a postulated leaking piping system would be small. Such small leakage, when compared with normal or standby flow capabilities of the systems, would not establish any appreciable dp across a closing isolation MOV until fully closed.

Further, there have been some inadvertent isolations of these MOVs over the years at operating plants. Some of these isolations have occurred at or near 100% system flow rates. This demonstrates isolation capability well in excess of small pipe leak flow conditions. It should be further noted that as the HPCI/RCIC valves close they are subjected to the full reactor pressure, (dp of 1000 psi) across the valve seat. This dp will be equivalent to the isolation MOV end of stroke dp conditions for a DEGB. Therefore, in-situ valve closure capability has been demonstrated. Successful RWCU isolations during normal full-flow operation have occurred, which subjects the valves to full reactor pressure (dp of 1000 psi) across the valve seat. Therefore, in-situ valve closure capability has been demonstrated. MOV isolation operability for small pipe leaks has been demonstrated for all three systems.

4.0 Safety Assessment - Design Basis Pipe Break

4.1 Realistic Analysis Conditions

An analytical assessment of a postulated design basis pipe break condition in one of the three BWR systems of concern can be looked at from a realistic perspective, just like the postulated small leak condition. A realistic review, without all of the design basis assumptions, was conducted because of the low probability (4×10^{-4}/yr) of a high energy line break in one of these systems. Any MOVs at BWRs which might be considered marginal or inadequate, when comparing their actuator size and deliverable stem force against expected required thrust, could still be instrumental in achieving system isolation.

Some beneficial conclusions can be drawn from the system design, equipment design, and physical attributes of the systems and equipment. There are MOV design considerations which have been included during the design process which make MOV actuators more capable than their ratings state.

The actual flow during a postulated leak would probably be closer to the 100% system flow rate rather than that attributable to the DEGB. This is because ductile pipe lines do not physically guillotine rupture and there would be a flow interference from the remaining piping. Some plant valves have already demonstrated the ability to close under comparable, full flow conditions when inadvertent system initiation and isolations have occurred.

There are two MOV isolation valves in series on each of these system supply lines. They are typically mounted in the supply lines very close to one another, separated only by the containment wall. Upon receipt of isolation signals they will not close at exactly the same time. This is because of real world, small physical differences, as well as the fact that some are driven by AC motors while others are driven by DC motors. Therefore, each valve may be subjected to different dp levels as they are closing. The possible alternate sharing of the break flow high pressure conditions and any cycling of this sharing between the two valves would probably allow at least one of the isolation valves to continue its closure motion until it becomes fully closed with the possibility of the second valve following thereafter. This possibility might better be described as a sharing or splitting of the high pressure condition between the valves. As the valves reach the end of stroke, they will be subjected to the full dp condition. However, as discussed in Section 3.7, this is equivalent to the conditions that these valves would experience at the end of travel during inadvertent isolation.

The control circuits for most MOVs contain limit switches for end of travel control, torque switches for valve seating (closing) control, and motor thermal overloads. These controls all have the potential to stop actuator travel. In some plants the typical control arrangement has the limit switch bypassing the torque switch for 95% of the valve closure stroke. The torque switch controls only during the final 5% of the valve closure stroke. Thus full actuator torque capability is available until after valve orifice closure. In addition, many MOVs have the motor thermal overload bypassed except for testing.

A full HPCI steam line break will reduce the reactor pressure. Therefore, the resulting dp loads on the valves will decrease with time during an outside containment line break event.

Even if the isolation valves are not fully closed, the operator will be aware that the break has not been isolated due to the break detection system alarm in the control room. Control room operator response to the existing Emergency Procedure Guidelines will lead quickly to reactor scram and depressurization. Once initiated, reactor depressurization occurs in a few minutes. Reactor system depressurization through the break and through automatic or manual actions will reduce the dp on the valve. This will allow time to isolate the line and ensure adequate core cooling.

The combination of the above factors leads to the conclusion that isolation MOVs will most likely respond to an intermediate pipe break condition or a design basis event with successful isolation.

4.2 Nuclear System Impact

Assuming the high energy line break occurs, external to the containment, in one of the three systems, the impact on the nuclear system would be less severe than a Design Basis Accident (DBA). The high energy lines are small lines (compared to the DBA) and would require less Emergency Core Cooling Systems (ECCS) flow for core cooling. Any one of the low pressure injection pumps (Core Spray or Low Pressure Coolant Injection) would be sufficient to provide core cooling and handle the consequences of a postulated line break. Existing SAR analyses for the same line breaks inside the containment (which cannot be isolated) show that there will not be any resulting core or fuel damage for the smaller line break events.

ECCS components have spatial separation such that the impact of the postulated high energy line break should affect only one division of equipment. The remaining division will be more than sufficient to handle even the maximum line break considered in this analysis (as opposed to a more likely small leak in the line).

Therefore, BWR plants have adequate safety margin to protect the reactor core and provide adequate leak detection and isolation capability using the presently designed isolation MOVs and other mitigating measures.

4.3 Offsite Dose Release Impact

The radiological release from the DEGB of the HPCI and RCIC steam line is bounded by that of the main steam line break. These smaller lines do not depressurize the reactor vessel as fast as the main steam line. The reactor inventory release for these breaks is mostly steam. The dose from steam loss through an outside line break is small. Therefore, the offsite release from the HPCI and RCIC steam line break will still meet requirements of 10CFR100. The reactor inventory loss from the DEGB of the RWCU line will be mostly liquid. However, the radiological consequences of the RWCU line is bounded by that of the main steam line, based on the assumed valve closure times for the RWCU isolation valves. The radiological release from the main steam line is only a small fraction of that of 10CFR100. Therefore, any slightly longer valve stroke time for the RWCU isolation valves will not result in noncompliance with the requirements of 10CFR100.

5.0 Conclusions

Because of the leak-before-break considerations for the HPCI/RCIC/RWCU piping, it is not expected that system isolation MOVs would ever be challenged at high flow design basis accident conditions. With the effective isolation systems, leaks should be isolated early at low flow conditions. Additionally, realistic consideration of expected plant and system response to postulated accident conditions leads to the conclusion that there is a significantly high probability of successful valve closure. Even without successful full valve closure for a postulated rupture in these lines, there is adequate safety margin in the ECCS to handle the reactor coolant inventory losses. The ECCS are designed for a much larger break than these small line ruptures. Delayed isolation response for these three systems is expected to keep offsite dose releases within 10CFR100 requirements.

6.0 References

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[7] 10 Code of Federal Regulations 50 Appendix A General Design Criteria

[8] NRC Generic Letter 88-01 NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, dated January 25, 1988.

[9] NRC IE Bulletin 85-03 Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings, dated November 15, 1985

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