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

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IMPORTANT NOTICE REGARDING
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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) of these MOVs has been performed to pipe break harsh environment envelope bounding conditions, which are much worse than small leak environmental conditions. Satisfaction of EQ requirements assures continued equipment safety function performance including MOVs up to these EQ bounding conditions. Therefore, no EQ concern exists for MOV isolation or the functioning of other safety systems equipment due to small pipe leaks.

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

[1] Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, NUREG-1061, Volumes 1 through 5, 1984.

[2] Federal Register, Volume 52, p. 41288, final rule modifying General Design Criterion 4 in 10 CFR Part 50, Appendix A.

[3] GESSAR II, 238 Nuclear Island, Section 5.2.5, GE Document No. 22A7007, Rev. 0.

[4] S. Ranganath and H. S. Mehta, "Engineering Methods for the Assessment of Ductile Fracture Margin in Nuclear Power Plant Piping," ASTM STP 803, 1983, pp. II-309 to II330.

[5] A. Zahoor, R.M. Gamble, H.S. Mehta, S. Yukawa and S. Ranganath, "Evaluation of Flaws in Carbon Steel Piping: Appendixes A and B," EPRI Report No. NP-4824SP, October 1986.

[6] Mehta, H.S., "Determination of Crack Leakage Rates in BWRs," Attachment 2 in Letter dated April 22, 1985, from Jack Fox, Chairman, ANS-58.2 Working Group to K. Wichman of NRC.

[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

TABLE 1

VALUES OF PARAMETERS USED IN CRITICAL CRACK LENGTH AND LEAK RATE CALCULATIONS

Pipe Thickness	:	Schedule 80
Pipe Internal Pressure	:	1050 psi
Temperature	:	528 °F
Normal Operation Bending Stresses	:	4 ksi
Material	:	Stainless Steel or Carbon Steel

TABLE 2

CRITICAL CRACK LENGTHS AND LEAK RATES FOR VARIOUS DIAMETER PIPES

Pipe Diameter (in.)	Critical Crack Length (in.)	Leak Rate at Critical Crack Length (gpm)	
		Water	Steam
4	7.1	25	15
6	9.8	41	27
12	18.5	166	108
16	23.1	262	170

RESPONSE TO
SECTION IV.B, CONTENTS OF PACKAGES SUBMITTED TO CRGR,
IN THE CRGR CHARTER (REVISION 4, APRIL 1987)

(i) The proposed generic requirement or staff position as it is proposed to be sent out to licensees.

The staff position is provided in proposed Supplement 3 to Generic Letter 89-10.

(ii) Draft staff papers or other underlying staff documents supporting the requirements or staff positions.

The NRC staff and the BWR Owners' Group safety assessments of the potential MOV deficiencies are provided as Enclosures 3 and 4, respectively, of this CRGR package. In addition, the staff will discuss its review of the NRC-sponsored test results, as summarized in Information Notice 90-40 (June 5, 1990), and the MOV data provided by the BWR Owners' Group at the CRGR briefing.

(iii) Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff positions, implement existing requirements or staff positions, or would relax or reduce existing requirements or staff positions.

Proposed Supplement 3 to Generic Letter 89-10 would increase the current staff position in that the supplement will result in advancing the recommended schedule in Generic Letter 89-10 for the BWR licensees to evaluate the capability of the MOVs used for containment isolation in the steam lines of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, in the supply line to the Reactor Water Cleanup (RWCU) system, and in the line to the isolation condenser, as applicable.

(iv) The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

The staff proposes to implement this staff position through the issuance of a supplement to Generic Letter 89-10. OGC has concurred in this CRGR submittal package.

(v) Regulatory analyses generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568.

A regulatory analysis as well as a value-impact analysis (NUREG/CR-5140) were prepared to support the staff's proposal to issue Generic Letter 89-10. The staff considers those analyses to be generally applicable to proposed Supplement 3 to Generic Letter 89-10. For the valves covered by this action, the staff believes that the failure rate may be significantly higher than the rate used in the Generic Letter 89-10 regulatory analysis and, accordingly, that the proposed advancement for the schedule is warranted. More directly, however, staff experts in risk assessment participated in the evaluation of the safety significance of the potential MOV deficiencies and the need to take action to ensure the capability of the MOVs covered by Supplement 3 in advance of the Generic Letter 89-10 schedule. Based on this information, the staff has determined that licensees may be allowed at least one refueling cycle to correct any MOV deficiencies.

(vi) Identification of the category of reactor plants to which the generic requirement or staff position is to apply.

The staff has limited the scope of the reporting requirements in proposed Supplement 3 to Generic Letter 89-10 to BWR licensees. In determining the MOVs to be evaluated under Generic Issue 87, RES found that the steam lines of the HPCI and RCIC systems and the supply line to the RWCU systems relied on the closure of MOVs to prevent the release of reactor coolant outside containment. Consequently, these MOVs are of high importance in mitigating offsite consequences of a pipe break. The NRC-sponsored tests involved valve types and sizes typically used in those applications and the conditions under which they are designed to operate. Based on the results of those tests, the staff has concluded that it has sufficient evidence to request BWR licensees to advance the schedule for evaluating the capability of the MOVs within the scope of Generic Issue 87. The information from these tests will likely be useful in determining the thrust requirements for MOVs in other systems in both BWRs and PWRs. The staff will expect all licensees to consider the MOV test results, where applicable, during the implementation of programs in response to Generic Letter 89-10.

(vii) For each such category or reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

(a) Statement of the specific objectives that the proposed action is designed to achieve;

As part of the effort to resolve Generic Issue 87, the NRC sponsored tests of valves typically used for containment isolation in the steam supply lines of the HPCI and RCIC systems, and in the supply line to the RWCU system at BWR plants. Those tests have revealed that the thrust required to close the tested valves under blowdown conditions was greater than previously predicted. From a comparison of the test results to the current capability of the MOVs used for those functions at BWR plants, the staff found that some of those MOVs appeared to be set or sized significantly below the thrust required during the tests. The objective of this proposed staff action is to identify and to correct deficiencies that might be present in the applicable MOVs in the HPCI, RCIC, and RWCU systems (and in lines to isolation condensers, as applibable) in advance of the schedule recommended in Generic Letter 89-10.

(b) General description of the activity that would be required by the licensee or applicant in order to complete the action;

In response to Supplement 3 to Generic Letter 89-10, BWR licensees would prepare a plant-specific safety assessment to verify that the safety assessments performed by the BWR Owners' Group and the NRC staff are applicable. BWR licensees would also need to evaluate the applicable MOVs in the HPCI, RCIC, and RWCU systems (and in lines to isolation condensers, as applicable) to determine whether deficiencies exist in their capability to perform their design-basis functions and to establish a schedule for the correction of identified deficiencies. Based on its safety assessment and review of the BWR Owners' Group safety assessment, the staff has concluded that justification exists for the continued operation of BWR plants for at least one refueling cycle even though potential deficiencies might exist in the MOVs covered by Supplement 3 to the generic letter. BWR licensees would need to justify any corrective action schedule that is longer than one refueling cycle. BWR licensees could perform the MOV evaluation as part of an advanced response to Generic Letter 89-10 for the specific MOVs in question.

With respect to the reporting requirements of the proposed Supplement 3, the BWR licensees will notify the staff of the availability on site of a plant-specific safety assessment within 30 days. The staff may conduct sample reviews of those plant-specific safety assessments. Within 90 days, BWR licensees will provide the criteria used to determine whether deficiencies exist in the applicable MOVs, will identify deficient MOVs, and will develop a schedule for any necessary corrective action. (The BWR licensees will also provide any subsequent changes to this information.) The staff will use this information to determine the safety significance of the MOV deficiencies and the adequacy of the schedule for any corrective action. The staff considers the potential significance of the MOV deficiencies to justify these reporting requirements.

(c) Potential change in the risk to the public from the accidental offsite release of radioactive material;

In the value-impact analysis performed for Generic Letter 89-10, the staff found that the net averted public risk and occupational exposure justified the issuance of the generic letter. The staff considers this analysis to be generally applicable to the proposed Supplement 3. Estimates made by the staff experts in risk assessment indicate that a reduction in risk would result from the correction of any deficiencies in the MOVs covered by Supplement 3.

(d) Potential impact on radiological exposure of facility employees;

The value-impact analysis performed for Generic Letter 89-10 indicated that the implementation of Generic Letter 89-10 will result in the avoidance of accidental and operational doses. The staff considers this determination to be applicable to proposed Supplement 3.

(e) Installation and continuing costs associated with the action, including the cost to facility downtime or the cost of construction delay;

The value-impact analysis for Generic Letter 89-10 indicated that an overall net cost benefit would result from implementation of the generic letter. The staff considers this determination to be applicable to proposed

Supplement 3. Nevertheless, to comply with the reporting requirements of the proposed supplement, the BWR licensee will need to perform tasks that will involve the expenditure of resources that might have been scheduled for a later date under the Generic Letter 89-10 program. The staff considers this early resource allocation to be justified to resolve the potential MOV deficiencies.

(f) The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements and staff positions;

The staff anticipates no changes in the physical complexity of the plant. Plant modifications might be needed to replace certain motors, actuators, or valves found inadequate. Operational procedures might be prepared to provide for operator action if the licensee identifies deficiencies in the MOVs. For example, such procedures might involve action to be taken in anticipation of a large pipe break based on leak detection capabilities.

(g) The estimated resource burden on the NRC associated with the proposed action and the availability of such resources;

The staff expects the implementation of Supplement 3 to involve a small resource burden. The staff will need to evaluate licensee action where deficient MOVs are found. This staff activity would have been necessary as part of the implementation of Generic Letter 89-10 at a later date.

(h) The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed action;

Because the reporting requirements in proposed Supplement 3 are limited to BWR licensees, those licensees will be expected to take action in the near term. PWR licensees will need to evaluate the applicability of the test data as part of their programs in response to Generic Letter 89-10. Older BWR plants might have a greater number of MOV deficiencies because 6 of these facilities have MOVs in similar applications in their isolation condenser lines.

(i) Whether the proposed action is interim or final and, if interim, the justification for imposing the proposed backfit on an interim basis.

If a BWR licensee is unable to justify the method used to verify that the MOV does not have a degraded condition, the licensee might set the MOV switches based on the best data currently available. In that event, the licensee will need to verify its selection of torque switch settings at a later date. As suggested by Generic Letter 89-10 and Supplement 1, licensees will likely use this "two stage approach" for many MOVs in the program. Therefore, such an approach is not unique to the MOVs covered by the proposed Supplement 3.

(viii) For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director's determination, together with the rationale for the determination based on the considerations of paragraphs (i) through (vii) above, that (a) there is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and (b) the direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

In proposing the issuance of Generic Letter 89-10, the staff determined that the implementation of this generic letter would substantially increase the overall protection of the public health and safety and would reduce overall costs to plant owners. The staff considers that this conclusion also applies to the proposed Supplement 3 to Generic Letter 89-10. The staff has concluded that the potential inadequacy of a substantial number of these valves justifies the verification of the capability of the MOVs covered by proposed Supplement 3 at this time.

(ix) For each evaluation conducted for proposed relaxations or decreases in current requirements or staff positions, the proposing office director's determination, ...

Item (ix) does not apply to this proposed staff action.

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

October 16, 1990

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 191

The Committee to Review Generic Requirements (CRGR) met on Friday, September 14, 1990 from 10:00 a.m.-3:00 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting:

1. J. Richardson, L. B. Marsh, E. Sullivan and T. Scarborough of NRR presented for CRGR review a proposed Supplement 3 to Generic Letter 89-10 on motor operated valves. The supplement would request that licensees consider problems found in NRC sponsored tests of certain valves and address any affected valves on a priority basis within the overall MOV testing program. The Committee supported the concept of requesting expedited action and provided a number of comments. The staff agreed to provide a redrafted letter for CRGR review. The CRGR review would be completed by negative consent, if possible. This matter is discussed in Enclosure 2.

2. R. Bosnak and J. Vora of RES and W. Travers, J. Craig and J. Thoma of NRR presented for CRGR review a proposed regulatory guide on standard format and content for license renewal and a proposed standard review plan for license renewal. The Committee recommended in favor of the proposed documents. The Committee provided a number of comments which the staff agreed to consider. No coordination with the CRGR staff or re-review by the CRGR was requested. This matter is discussed in Enclosure 3.

In accordance with the EDO's July 18, 1983 directive concerning "Feedback and Closure of CRGR Reviews," a written response is required from the cognizant office to report agreement or disagreement with CRGR recommendations in these minutes. The response, which is required within five working days after receipt of these minutes, is to be forwarded to the CRGR Chairman and if there is disagreement with CRGR recommendations, to the EDO for decisionmaking.

~~9010260222 XA~~

Questions concerning these meeting minutes should be referred to Dennis Allison (492-4148).

Original Signed by:
E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures:
As stated

cc w/enclosures:
Commission (5)
SECY
J. Lieberman
P. Norry
D. Williams
Regional Administrators
CRGR Members

Distribution:
Central File (w/o encl.)
PDR/DCS (NRC/CRGR) (w/o encl.)

P. Kadambi	CRGR CF
CRGR SF	J. Sniezek
M. Taylor	J. Heltemes
J. Craig	J. Richardson
L. Marsh	E. Sullivan
T. Scarborough	R. Bosnak
L. Shao	J. Varga
J. Thoma	D. Ross
E. Jordan	J. Conran
D. Allison	

OFC	:CRGR/AEOD	:DD:AEOD	:C/CRGR:AEOD	:	:	:	:
NAME	:DAllison:sm	:DRoss	:EJordan	:	:	:	:
DATE	:10/16/90	:10/16/90	:10/16/90	:	:	:	:

OFFICIAL RECORD COPY

Enclosure 1

ATTENDANCE LIST

CRGR Meeting No. 191

September 14, 1990

CRGR Members

E. Jordan
G. Arlotto
J. Moore
F. Miraglia
B. Sheron
L. Reyes

CRGR Staff

J. Conran
D. Allison

NRC Staff

W. Minners
J. Richardson
L. Marsh
E. Sullivan
T. Scarborough
R. Bosnak
J. Vora
J. Craig
J. Thoma
M. Davis
M. Taylor
C. Thompson
F. Gurbelich
A. Gody
G. Weidenhamer
R. Kiessel
O. Rothberg
F. Akstulewicz
G. Mizuno
P. T. Kuo
R. Anand
R. Borchardt
D. Jackson
E. Doolittle
J. Murphy

Enclosure 2 to the Minutes of CRGR Meeting No. 191
Proposed Supplement 3 to Generic Letter 89-10
on Motor Operated Valve Testing

September 14, 1990

TOPIC

J. Richardson, L. B. Marsh, E. Sullivan and T. Scarborough presented for CRGR review a proposed Supplement 3 to Generic Letter 89-10 on Motor Operated Valves. The supplement would request that licensees consider problems found in NRC-sponsored tests of certain valves and address any affected valves on a priority basis within the overall MOV testing program that was being conducted under Generic Letter 89-10.

In pursuing resolution of Generic Issue 87, "Failure of HPCI Without Isolation," the NRC has sponsored tests on 6-and 10-inch gate valves typically used to perform containment isolation in the steam supply lines to HPCI and RCIC systems and in the water supply line to the RWCU system in BWR's. The results indicated that the thrust required to close the valves under blowdown conditions associated with a pipe break was greater than previously predicted. Because of the important function of these valves, the staff was proposing that BWR licensees determine the applicability of this information to valves in their plant and take expedited actions for any deficiencies found. In addition, because the mechanisms involved, such as under predicting friction factors, could apply widely, all licensees would be requested to assess the applicability of this information to other valves in their plants.

The slides used by the staff in the presentation are provided as an attachment to this enclosure.

BACKGROUND

The review packaged was forwarded by a memorandum dated August 31, 1990 from F. Miraglia to E. Jordan. The package included:

(1) Proposed supplement.

(2) Memorandum dated August 13, 1990 from J. Richardson to W. Russell, Subject: Safety Concern Relative to BWR Containment Isolation Valves for HPCI, RCIC and RWCU.

(3) Letter dated July 27, 1990 from G. Beck, BWR Owners' Group, to J. Richardson, NRC, Subject: BWR Owners' Group Safety Assessment of MOV Isolation Function.

(4) Responses to contents of packages submitted for CRGR review.

CONCLUSIONS/RECOMMENDATIONS

The CRGR supported the concept of requesting expedited action within the context of the overall MOV testing program, and provided a number of comments. The staff agreed to provide a redrafted letter for CRGR review. If possible the CRGR review would be completed by negative consent rather than at another meeting.

The following suggestions were made:

(1) BWR licensees should be requested:

(a) to describe their findings and plans with respect to these particular valves (e.g., complete the valve testing program within 18 months or, justify the extended time).

(b) to address the applicability of the information developed in the NRC-sponsored tests to other valves determine the priorities for their entire valve testing programs under Generic Letter 89-10.

(2) PWR licensees should also consider the applicability of the information obtained from the MOV tests and the staff's safety evaluation to other MOV's. However, the reporting requirements of the supplement should be addressed to BWR's only.

(3) The backgound discussion should be expanded further to discuss the friction factor problem and how it may apply to other sizes and models of valves. It should also indicate the desirability of a final fix instead of a temporary fix. It might, in some cases, take longer than 18 months to achieve a final fix.

(4) Licensees should be requested to implement appropriate procedures pending completion of any corrective actions on the valves.

(5) The basis for the letter should be compliance rather than adequate protection. The staff should confirm this aspect with OGC.

This action was considered to be a justified backfit, within the compliance exception in the backfit rule.

Safety goal considerations were not discussed at this meeting.

DISCUSSION OF

THE PROPOSED SUPPLEMENT 3 TO GENERIC LETTER 89-10

WITH THE COMMITTEE TO REVIEW GENERIC REQUIREMENTS

September 14, 1990

Attachment to Enclosure 2

GENERIC LETTER 89-10
SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE

ISSUED JUNE 28, 1989

REQUESTED LICENSEES TO ESTABLISH PROGRAMS TO ENSURE CAPABILITY OF ALL MOVs IN SAFETY-RELATED SYSTEMS TO PERFORM THEIR SAFETY FUNCTION.

RECOMMENDS TESTING OF MOVs AT DESIGN-BASIS DIFFERENTIAL PRESSURE AND FLOW CONDITIONS WHERE PRACTICABLE. ALTERNATIVES MAY BE USED WHERE DESIGN-BASIS TESTING NOT PRACTICABLE.

OUTLINES "TWO STAGE" APPROACH FOR INSTANCES WHERE DESIGN-BASIS TESTING NOT PRACTICABLE AND AN ALTERNATIVE CANNOT BE JUSTIFIED AT THIS TIME. WITH THE TWO STAGE APPROACH, MOV SWITCH SETTINGS DETERMINED USING THE BEST DATA AVAILABLE AND THEN LICENSEE OBTAINS APPLICABLE DATA AS SOON AS POSSIBLE.

REQUESTS PERIODIC VERIFICATION OF MOV SWITCH SETTINGS EVERY 5 YEARS.

LICENSEES WERE REQUIRED TO INDICATE THEIR INTENTIONS BY DECEMBER 28, 1989.

PROPOSED SCHEDULE REQUESTED PROGRAM DESCRIPTION ONSITE BY JUNE 28, 1990, OR FIRST REFUELING OUTAGE AFTER DECEMBER 28, 1989, WHICHEVER WAS LATER. (MODIFIED IN SUPPLEMENT 2)

PROPOSED SCHEDULE REQUESTS COMPLETION OF INITIAL TEST PROGRAM BY JUNE 1994 OR 3 REFUELING OUTAGES AFTER DECEMBER 28, 1989, WHICHEVER IS LATER.

GENERIC LETTER 89-10
(continued)

JUNE 13, 1990 SUPPLEMENT 1

PROVIDES THE RESULTS OF THE PUBLIC WORKSHOPS TO DISCUSS THE GENERIC LETTER AND TO ANSWER QUESTIONS REGARDING ITS IMPLEMENTATION.

LIMITS SCOPE OF GENERIC LETTER TO MOVs IN SAFETY-RELATED <u>PIPING</u> SYSTEMS.

LIMITS CONSIDERATION OF VALVE MISPOSITIONING TO INADVERTENT OPERATION FROM THE CONTROL ROOM.

DISCUSSES THE FACTORS TO BE CONSIDERED, AND LIMITATIONS, IN JUSTIFYING THE ACCEPTABILITY OF ALTERNATIVES TO TESTING MOVs IN SITU UNDER DESIGN-BASIS DIFFERENTIAL PRESSURE AND FLOW CONDITIONS.

EMPHASIZES THE RECOMMENDATION TO FOLLOW THE TWO STAGE APPROACH WHERE DESIGN-BASIS TESTING IS NOT PRACTICABLE AND AN ALTERNATIVE CANNOT BE JUSTIFIED AT THIS TIME.

AUGUST 3, 1990 SUPPLEMENT 2

ALLOWS ADDITIONAL TIME FOR LICENSEES TO INCORPORATE THE INFORMATION IN SUPPLEMENT 1 INTO THEIR GENERIC LETTER PROGRAMS BY STATING THAT PROGRAM DESCRIPTIONS NEED NOT BE AVAILABLE ON SITE UNTIL JANUARY 1, 1991.

GENERIC ISSUE 87
FAILURE OF HPCI STEAM LINE WITHOUT ISOLATION

INITIAL SCOPE: CONTAINMENT ISOLATION MOTOR-OPERATED GATE VALVES IN HPCI AND RCIC STEAM TURBINE LINES, AND RWCU SUPPLY LINE.

PHASE I (1988) TESTING: 2 SIX-INCH RWCU VALVES (ANCHOR/DARLING AND VELAN) UNDER HIGH ENERGY HOT WATER LOADS.

PHASE II (1989) TESTING: 3 SIX-INCH RWCU VALVES (ANCHOR/DARLING, VELAN, AND WALWORTH) AND 3 TEN-INCH HPCI VALVES (ANCHOR/DARLING, POWELL, AND VELAN) UNDER NORMAL AND BLOWDOWN LOADS.

PUBLIC MEETINGS ON FEBRUARY 1, 1989 AND APRIL 18, 1990.

INFORMATION NOTICE 90-40 (JUNE 5, 1990), RESULTS OF NRC-SPONSORED TESTING OF MOTOR-OPERATED VALVES

1. MORE THRUST REQUIRED THAN PREDICTED FROM STANDARD INDUSTRY EQUATION
2. SOME VALVES INTERNALLY DAMAGED AND REFERRED TO AS "UNPREDICTABLE"
3. STATIC AND LOW FLOW TESTING MIGHT NOT PREDICT PERFORMANCE UNDER DESIGN-BASIS FLOW CONDITIONS
4. DURING OPENING, HIGHEST LOAD NOT ALWAYS AT UNSEATING
5. PARTIAL STROKING DID NOT REVEAL REQUIRED THRUST
6. TORQUE, THRUST, AND MOTOR OPERATING PARAMETERS NEEDED TO FULLY CHARACTERIZE MOV PERFORMANCE
7. RELIABLE USE OF MOV DIAGNOSTICS NEEDS ACCURATE EQUIPMENT AND TRAINED PERSONNEL.

STAFF EVALUATION OF THE MOV TEST RESULTS

ON MAY 10, 1990, NRC VALVE REVIEW GROUP MET TO DISCUSS THE NEED FOR PROMPT STAFF ACTION IN RESPONSE TO THE MOV TEST RESULTS.

AFTER DISCUSSIONS WITH NRR MANAGEMENT, STAFF CONDUCTED INFORMAL SURVEY OF 6 BWR UNITS TO DETERMINE THE CAPABILITY OF THE MOVs USED FOR CONTAINMENT ISOLATION IN THE STEAM LINE OF THE HPCI AND RCIC SYSTEMS, AND IN THE SUPPLY LINE FOR THE RWCU SYSTEM.

ON MAY 24, STAFF MET WITH BWR OWNERS GROUP TO DISCUSS THE RESULTS OF THAT SURVEY.

IN RESPONSE TO STAFF CONCERNS, THE BWR OWNERS GROUP AGREED TO OBTAIN SIMILAR INFORMATION FOR THE REMAINING BWR UNITS.

ON JULY 6, THE BWR OWNERS GROUP PROVIDED INFORMATION ON THE CURRENT CAPABILITY OF MOVs USED FOR CONTAINMENT ISOLATION IN THE HPCI, RCIC AND RWCU SYSTEMS.

AFTER EVALUATING THE PROVIDED INFORMATION, THE STAFF ACTIVATED THE BWR REGULATORY RESPONSE GROUP (RRG). PUBLIC MEETINGS WERE THEN HELD ON AUGUST 1 AND SEPTEMBER 7 TO DISCUSS SAFETY ASSESSMENTS PERFORMED BY THE STAFF AND THE BWR OWNERS GROUP, AND ACTIONS PLANNED BY THE STAFF AND THE BWR OWNERS GROUP.

MOV DATA REQUESTED FROM THE BWR OWNERS GROUP

FOR THE MOVs USED FOR 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 TO THE REACTOR WATER CLEANUP (RWCU) SYSTEM, THE FOLLOWING DATA WERE REQUESTED:

1. TYPE AND SIZE OF MOTOR, ACTUATOR, AND VALVE (INCLUDING DISK),

2. MANUFACTURER OF MOTOR, ACTUATOR, AND VALVE,

3. DESIGN DIFFERENTIAL PRESSURE AND FLUID TEMPERATURE FOR OPENING AND CLOSING OF THE VALVE, AND

4. THRUST DELIVERED AT THE CURRENT TORQUE SWITCH SETTING, DIFFERENTIAL PRESSURE AT WHICH TESTS CONDUCTED, AND BASIS FOR DELIVERED THRUST VALUE.

METHODOLOGY USED IN THE EVALUATION OF THE MOV DATA

1. EVALUATE ONLY GATE VALVES (GLOBE VALVES ASSUMED TO BE ADEQUATE).

2. FOR GATE VALVES WITH SAME SIZE AND MANUFACTURER AS THOSE IN NRC TESTS, USE INFORMATION NOTICE 90-40 TO ESTIMATE THRUST REQUIREMENTS.

3. FOR GATE VALVES WITH SAME SIZE BUT DIFFERENT MANUFACTURER FROM NRC TESTS, ASSUME THE VALVE PERFORMS IN A MANNER SIMILAR TO TESTED VALVES REQUIRING THE LEAST AMOUNT OF THRUST AMONG THOSE TESTED FOR THE SAME FLUID CONDITIONS.

4. FOR GATE VALVES WITH DIFFERENT SIZE THAN TESTED VALVES, THE THRUST REQUIREMENTS WERE ESTIMATED ASSUMING THE VALVE WAS NOT DAMAGED DURING OPERATION.

5. TORQUE SWITCH SETTINGS FOR EACH GATE VALVE IDENTIFIED BY THE BWROG WERE COMPARED TO ESTIMATED THRUST REQUIREMENTS.

6. ACTUATOR RATINGS WERE COMPARED TO ESTIMATED THRUST REQUIREMENTS.

7. MOTOR SIZES WERE COMPARED TO MOTOR SIZES USED IN TESTS AND ESTIMATES OF MOTOR THRUST CAPABILITY.

7/31/90

BWROG MOV DATA OVERVIEW

HPCI	TOTAL NUMBER OF VALVES = 46	
MOVs WITHOUT IDENTIFIED CONCERNS (INCLUDING 4 GLOBE VALVES)		18
MOVs WITH MARGINAL MOTOR, ACTUATOR, OR T. S. SETTING		16
MOVs WITH SMALL (OR LOW) MOTOR, ACTUATOR, OR T. S. SETTING UNITS L, M, P, T, V, Z, HATCH 1, HATCH 2, MONTICELLO* (9 OUT OF 23 REACTOR UNITS) * JUSTIFICATION SUPPLIED		12

RCIC	TOTAL NUMBER OF VALVES = 62	
MOVs WITHOUT IDENTIFIED CONCERNS (INCLUDING 7 GLOBE VALVES)		47
MOVs WITH MARGINAL MOTOR, ACTUATOR, OR T. S. SETTING		9
MOVs WITH SMALL (OR LOW) MOTOR, ACTUATOR, OR T. S. SETTING UNITS E, G, N, Q, T (5 OUT OF 30 REACTOR UNITS)		6

RWCU	TOTAL NUMBER OF VALVES = 71	
MOVs WITHOUT IDENTIFIED CONCERNS (INCLUDING 8 GLOBE VALVES)		19
MOVs WITH MARGINAL MOTOR, ACTUATOR, OR T. S. SETTING		12
MOVs WITH SMALL (OR LOW) MOTOR, ACTUATOR, OR T. S. SETTING UNITS B, D, H, I, K, L, N, P, Q, R, S, T, U, V, W, Y, Z, AC, HATCH 2, QUAD CITIES 1, QUAD CITIES 2 (21 OUT OF 34 REACTOR UNITS)		40

8 UNITS WITH MOV PROBLEMS (SMALL/LOW CATEGORY) IN MULTIPLE SYSTEMS

HPCI + RCIC + RWCU	1 (T)
HPCI + RCIC	0
HPCI + RWCU	5 (L, P, V, Z, HATCH 2)
RCIC + RWCU	2 (N, Q)

7/31/90

EXAMPLES OF STAFF FINDINGS

UNIT	SYSTEM	VALVE	SIZE (in.)	D/P (psid)	T.S. SETTING (lbs)	THRUST ESTIMATE FROM TEST (lbs)
M	HPCI	CRANE	10	1200	17460	29000
M	HPCI	CRANE	10	1200	22540	29000
T	HPCI	A/D	10	1250	26271	30000
T	HPCI	A/D	10	1250	20326	30000
V	HPCI	CRANE	10	1250	24017	29000
HATCH 1	HPCI	CRANE	10	1080	23055	29000
Q	RCIC	A/D	10	1146	23478	30000
D	RWCU	A/D	6	1020	12300	20000
D	RWCU	A/D	6	1020	16100	20000
I	RWCU	A/D	6	1190	10039	20000
K	RWCU	A/D	6	1040	12241	20000
K	RWCU	A/D	6	1040	14928	20000
L	RWCU	A/D	6	1150	13233	20000
L	RWCU	A/D	6	1150	13220	20000
N	RWCU	A/D	6	1250	13405	20000
N	RWCU	A/D	6	1250	13405	20000
P	RWCU	A/D	6	1150	16069	20000
P	RWCU	A/D	6	1150	13786	20000
Q	RWCU	A/D	6	1250	13405	20000
Q	RWCU	A/D	6	1250	13405	20000
R	RWCU	A/D	6	1173	13780	20000
S	RWCU	A/D	6	1025	12800	20000
S	RWCU	A/D	6	1025	12800	20000
T	RWCU	A/D	6	1020	9354	20000
T	RWCU	A/D	6	1020	11465	20000
W	RWCU	A/D	6	1135	15400	20000
Y	RWCU	A/D	6	1025	12800	20000
Y	RWCU	A/D	6	1025	12800	20000
QC 1	RWCU	CRANE	6	1250	6506	12000
QC 1	RWCU	A/D	6	1250	8333	20000
QC 2	RWCU	CRANE	6	1250	4004	12000
QC 2	RWCU	A/D	6	1250	10190	20000

NRC STAFF SAFETY ASSESSMENT OF POTENTIAL MOV DEFICIENCIES IN HPCI, RCIC, AND RWCU SYSTEMS

LIKELIHOOD OF PIPE BREAK

HPCI and RCIC Low Erosion/Corrosion Susceptibility

HPCI and RCIC steam lines predicted to have insignificant erosion/corrosion.

RWCU Augmented Inspections

In response to GL 88-01, licensees have committed to augmented inspections of RWCU supply lines.

Piping Stress Levels

ASME Section III piping has substantial margin between allowable stress and material ultimate strength.

Failure Mechanisms

Large pipe breaks have low probability. Small break/leak likely to be detected by temperature and sump level monitors with early MOV closure by plant procedures.

PLANT MITIGATIVE FEATURES

Margin on Assumed Differential Pressure

Actual differential pressure during the blowdown event might be lower than design differential pressure.

Valve Redundancy

Partial closure of MOVs in series might reduce the pressure load on each MOV.

Closure After Depressurization

If not significantly damaged by unsuccessful closure attempt, MOV might be closed following depressurization.

Consequence Mitigation

If makeup water available, core cooling can continue with available systems until broken line is isolated.

RISK PROBABILITY ANALYSIS

Staff risk experts determined potential MOV deficiency should be resolved promptly, but immediate action not justified. Preliminary results of sensitivity analysis available by late October 90.

SUPPLEMENT 3 TO GENERIC LETTER 89-10

BACKGROUND and DISCUSSION

NRC-SPONSORED TESTS OF MOVS TYPICALLY USED TO PROVIDE CONTAINMENT ISOLATION IN STEAM LINES OF HPCI AND RCIC SYSTEMS AND IN THE SUPPLY LINE TO RWCU SYSTEMS REVEALED THAT THE THRUST REQUIRED TO CLOSE THE VALVES UNDER BLOWDOWN CONDITIONS WAS HIGHER THAN PREVIOUSLY PREDICTED. STAFF REVIEW OF NRC TEST DATA AND MOV DATA PROVIDED BY BWR LICENSEES INDICATES THAT MOV DEFICIENCIES MIGHT EXIST.

REQUESTED LICENSEE ACTIONS

BWR LICENSEES SHOULD ASSESS APPLICABILITY OF NRC TEST DATA; DETERMINE AS-IS CAPABILITY OF HPCI, RCIC AND RWCU MOVs; AND IDENTIFY MOV DEFICIENCIES.

BWR LICENSEES SHOULD PERFORM PLANT-SPECIFIC SAFETY ASSESSMENTS TO VERIFY STAFF AND BWROG ASSESSMENTS (ENCLOSURES TO SUPPLEMENT 3)

ALL LICENSEES SHOULD CONSIDER THE APPLICABILITY OF THE NRC TEST DATA IN THEIR GENERIC LETTER 89-10 PROGRAMS

REPORTING REQUIREMENTS

1. WITHIN 30 DAYS, BWR LICENSEES SHALL NOTIFY STAFF OF AVAILABILITY OF PLANT-SPECIFIC SAFETY ASSESSMENT.

2. WITHIN 90 DAYS, BWR LICENSEES SHALL PROVIDE
   (a) CRITERIA APPLIED IN DETERMINING WHETHER MOV DEFICIENCIES EXIST,
   (b) IDENTIFICATION OF DEFICIENT MOVs, AND
   (c) SCHEDULE FOR ANY NECESSARY CORRECTIVE ACTION.

3. BWR LICENSEES SHALL INFORM STAFF OF ANY CHANGES TO PLANNED ACTIONS OR SCHEDULE.

SUPPLEMENT 3 STATES THAT STAFF SAFETY ASSESSMENT JUSTIFIES CONTINUED OPERATION FOR AT LEAST ONE REFUELING CYCLE TO RESOLVE MOV DEFICIENCIES. BWR LICENSEES SHOULD PROVIDE JUSTIFICATION IF ADDITIONAL TIME NEEDED.

Enclosure 3 to the Minutes of CRGR Meeting No. 191
Proposed Regulatory Guide on Standard Format
and Content for Licensing Renewal and Proposed
Standard Review Plan for License Renewal

September 14, 1990

TOPIC

R. Bosnak and J. Vora of RES and W. Travers, J. Craig and J. Thoma of NRR presented for CRGR review a proposed regulatory guide on standard format and content for license renewal and a proposed standard review plan for license renewal. The documents were intended to be forwarded to the Executive Director for Operations and the Commission and then to be published for comment. They had been drafted to support a proposed rule (10 CFR 54) which had been published for comment on July 17, 1990. They generally implemented the provisions of the proposed rule. It was understood that, if the rule should change in a material way, the regulatory guide and standard review plan would also need to be changed. It was also recognized that the documents would be revised as the staff learned more about license renewal issues and their resolutions.

A copy of the slides used by the staff in the presentation is provided as an attachment to this enclosure.

BACKGROUND

The package provided for CRGR review was transmitted by a memorandum dated August 31, 1990 from E. Beckjord and T. Murley to E. Jordan. The package included:

1. Proposed regulatory guide on standard format and content.

2. Proposed standard review plan.

CONCLUSIONS/RECOMMENDATIONS

The Committee recommended in favor of the proposed documents. The Committee provided a number of comments which the staff agreed to consider. No coordination with the CRGR staff or re-review by the CRGR documents was requested.

The staff indicated in the review package that these documents were not considered backfits. The CRGR had no questions or comments on this determination.

The staff indicated in the presentation that the proposed documents were aimed at maintaining the current licensing basis during the renewal term and the relationship of a facility to the safety goals would, therefore, remain consistent with that of the original license term (see Slide 6). The CRGR had no questions or comments on this determination.

Draft Regulatory Guide DG-1009

and

Standard Review Plan for License Renewal (SRP-LR)

Presentation to CRGR
September 14, 1990

William D. Travers, John Craig, and John Thoma
Office of Nuclear Reactor Regulation

Robert J. Bosnak and Jitendra Vora
Office of Nuclear Regulatory Research

Attachment to
Enclosure 3

Purpose

* To discuss the draft R.G. on Format and Content.

* To discuss the draft Standard Review Plan for License Renewal.

* To request CRGR to make a positive recommendation to the EDO.

Presentation Outline

* Introductory Remarks
* Schedule
* Safety Goals
* Draft Regulatory Guide DG-1009
* Draft Standard Review Plan for License Renewal (SRP-LR)
* Respond to Questions

Introductory Remarks

License Renewal Involves Many Integrated Activities :

* Rulemaking (10 CFR Part 51 and 10 CFR Part 54)
* Regulatory Guide Development
* Standard Review Plan for License Renewal Development
* Industry Report Development and Review
* Lead Plant Reviews

Schedule for RG and SRP-LR

* Meet with CRGR in September 1990.
* Meet with the ACRS in October 1990.
* SRP-LR and R.G. to EDO by October 19, 1990.
* SRP-LR and R.G. to Commission by November 2, 1990.
* Publish for public comment by mid-December 1990.
* Revised package to ACRS/CRGR by November 1991.
* Revised package published by April 1992.

Safety Goals

* Atomic Energy Act allows provisions for license renewal (see 10 CFR 50.51).

* The actions and criteria described in the draft Regulatory Guide and Standard Review Plan for License Renewal provide guidance to the licensees and the staff.

* CLB Maintained

* Therefore, the relationship of the facility to the safety goals remains consistent with that of the original licensing term.

INTRODUCTORY REMARKS ON BACKGROUND OF REGULATORY GUIDE DEVELOPMENT

DISCUSSION OF NEEDED REGULATORY DOCUMENTS TO SUPPORT LICENSE RENEWAL RULE USING NPAR PROGRAM RESULTS (1987-89)

POSSIBLE REGULATORY GUIDE CANDIDATES (SECY-89-275)

- MAJOR COMPONENTS AND STRUCTURES
- SIGNIFICANT AGING MECHANISMS
- SELECTION OF COMPONENTS AND STRUCTURES
- FORMAT AND CONTENT OF TECHNICAL INFORMATION

DECISION REACHED (RES & NRR) IN 1989 TO DEVELOP SINGLE GUIDE ON FORMAT AND CONTENT OF TECHNICAL INFORMATION INCLUDING GUIDANCE ON AGING MANAGEMENT AND SCREENING (SECY 90-021)

AS REPORTED IN SECY 90-021, DECISION ANTICIPATED THAT INDUSTRY REPORT PROCESS BY NUMARC WILL PROVIDE FOR SPECIFIC COMPONENT NEEDS, AGING MECHANISMS, AND SCREENING. IF UNSUCCESSFUL, NEEDED RG/SRP WILL BE PREPARED AS REQUIRED.

<u>DRAFT R.G. DG-1009</u>

- STANDARD FORMAT AND CONTENT OF TECHNICAL INFORMATION FOR APPLICATION TO RENEW NUCLEAR POWER PLANT OPERATING LICENSES

- PURPOSE
- SCOPE
- FORMAT FOR TECHNICAL INFORMATION
- TECHNICAL INFORMATION CONTENT
  - SSC IMPORTANT TO LICENSE RENEWAL
  - SC REQUIRING EVALUATION OF AGE RELATED DEGRADATIONS
    - UNDERSTANDING AGING
      - AGING MECHANISMS
    - MANAGING AGING
  - RECORDKEEPING AND TRENDING
- REGULATORY ANALYSIS
- BACKFIT ANALYSIS

PURPOSE OF R.G. DG-1009

PROVIDE REGULATORY GUIDELINES FOR A UNIFORM FORMAT AND CONTENT FOR TECHNICAL INFORMATION TO BE SUBMITTED AS PART OF LICENSE RENEWAL APPLICATION

SCOPE

INCLUDES:

- FORMAT AND CONTENT OF TECHNICAL INFORMATION
- CRITERIA FOR SELECTION OF SYSTEMS, STRUCTURES, AND COMPONENTS (SSC) IMPORTANT TO LICENSE RENEWAL
- GUIDELINES FOR
  - UNDERSTANDING AGING
  - MANAGING AGING

FORMAT FOR TECHNICAL INFORMATION

- FORMAL APPLICATION
  - SUMMARY OF FINDINGS
  - IMPLEMENTATION PLAN

- FSAR SUPPLEMENTAL INFORMATION
  - SYSTEMS
  - COMPONENTS
  - SUPPORTING DOCUMENTATION

TECHNICAL INFORMATION CONTENT

PROVIDES GUIDELINES FOR:

- SELECTION OF SSC IMPORTANT TO LICENSE RENEWAL (ITLR)
- INTEGRATED PLANT ASSESSMENT
  - UNDERSTANDING AGING
  - MANAGING AGING
    - ESTABLISHED EFFECTIVE PROGRAMS
    - ACTIONS TO BE TAKEN

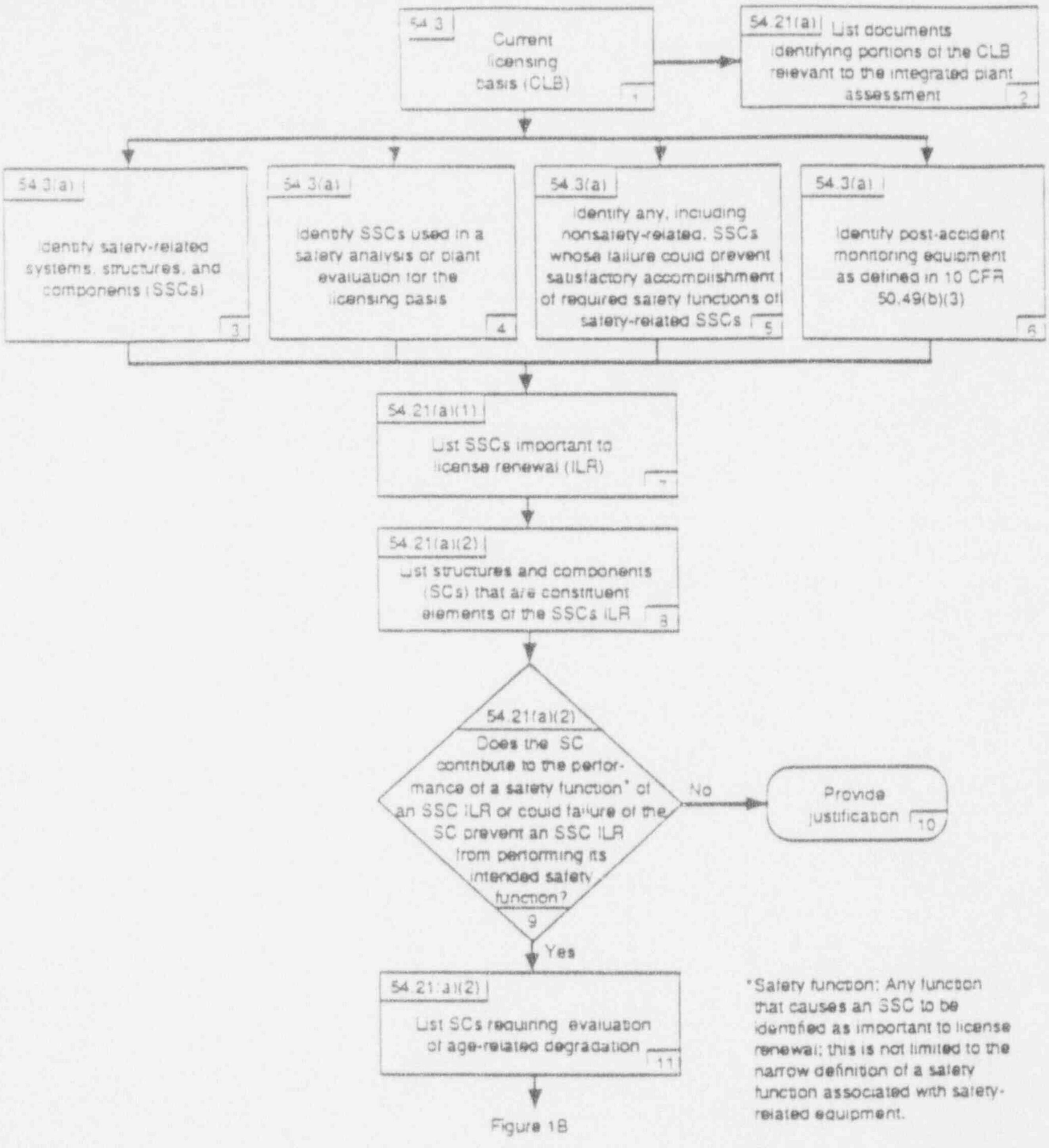

FIGURE 1A
Integrated Plant Assessment -- Identification of Important-To-License-Renewal SSCs and SCs Requiring Evaluation of Age-Related Degradation

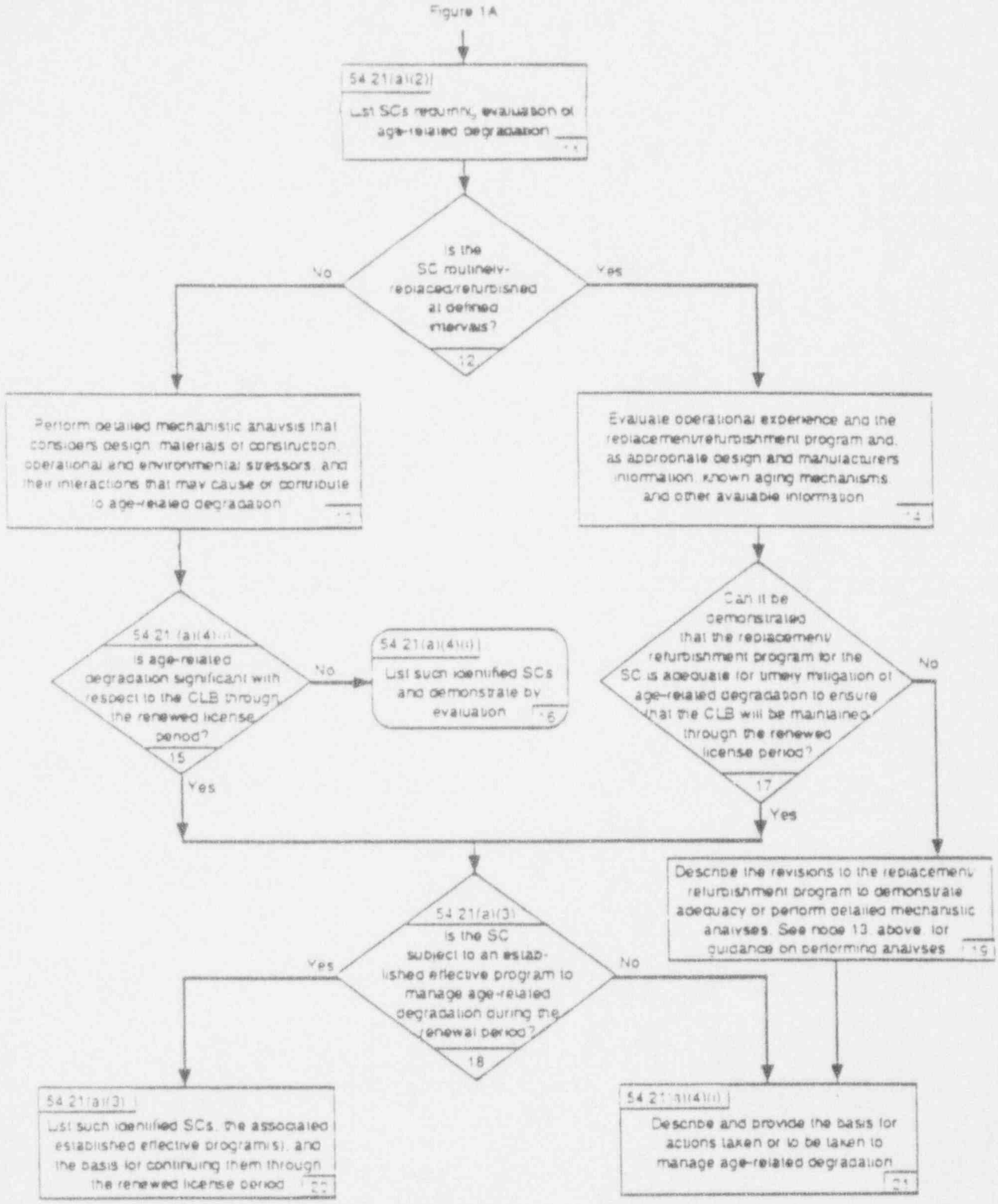

FIGURE 1B
Integrated Plant Assessment -- Evaluation of Age-Related Degradation

- RECORDKEEPING AND TRENDING

  10CFR54 REQUIREMENTS

  AUDITABILITY AND RETRIEVABILITY

  IMPLEMENTATION THROUGH PLANT PROGRAMS

  CONTROL OF ADMINISTRATIVE PROCESS

  APPLICATIONS FOR AGING MANAGEMENT

- REGULATORY ANALYSIS

  NUREG-1362 CONTAINS REGULATORY ANALYSIS FOR PROPOSED 10CFR54 RULE

- BACKFIT ANALYSIS

  NOT A BACKFIT UNDER 10CFR50.109

SINCE THE R.G. DG-1009 IMPLEMENTS 10CFR54 NO ADDITIONAL REGULATORY OR BACKFIT ANALYSIS IS NECESSARY

- SELECTION OF SSC IMPORTANT TO LICENSE RENEWAL
  - 10CFR54 REQUIREMENTS
  - DETERMINISTIC APPROACH
  - RISK-BASED SUPPLEMENTAL APPROACH

- UNDERSTANDING AGING

10CFR54 REQUIREMENTS

MATERIALS

STRESSORS

ENVIRONMENT

SERVICE CONDITION

MECHANISMS

DEGRADATION SITES

ROOT CAUSE(S)

- AGING MECHANISMS

  FATIGUE

  EROSION

  EROSION/CORROSION

  RADIATION EMBRITTLEMENT

  THERMAL EMBRITTLEMENT

  CORROSION

  WEAR

  SHRINKAGE/CREEP

  CHEMICAL EFFECTS/CONTAMINATION

- MANAGING AGING

10CFR54 REQUIREMENTS

INSPECTION

SURVEILLANCE

CONDITION MONITORING

NONDESTRUCTIVE EXAMINATION

ROOT CAUSE ANALYSIS

REPAIR, REFURBISHMENT

REPLACEMENT/CORRECTIVE MAINTENANCE

PREVENTIVE MAINTENANCE

PREDICTIVE MAINTENANCE

ADJUSTMENTS IN DESIGNS,

OPERATIONAL ENVIRONMENT

SERVICE CONDITIONS

STANDARD REVIEW PLAN FOR LICENSE RENEWAL (SRP-LR)

DRAFT NUREG 1299

* PURPOSE
* SCOPE
* ORGANIZATION
* REVIEW CRITERIA
* IMPLEMENTATION
* FUTURE REVISIONS

A DOCUMENT WHICH PROVIDES A FRAMEWORK FOR REVIEW AND WHICH WILL BE REVISED AS A RESULT OF PUBLIC COMMENTS AND AS EXPERIENCE IS GAINED FROM INDUSTRY TECHNICAL REPORTS, PILOT PLANT APPLICATION REVIEWS, AND ONGOING RESEARCH.

PURPOSE OF SRP-LR

* PROVIDE STAFF GUIDANCE FOR REVIEW OF THE:
  - SUFFICIENCY OF AN APPLICATION
  - APPLICANT'S SCREENING METHODOLOGY
  - POTENTIAL AGE-RELATED DEGRADATION MECHANISMS FROM A:
    + SYSTEM PERSPECTIVE
    + COMPONENT PERSPECTIVE
* PROVIDE GUIDANCE TO EVALUATE AGE-RELATED MANAGEMENT ACTIVITIES TO DETERMINE WHETHER OR NOT AN ESTABLISHED EFFECTIVE PROGRAM WILL BE OR HAS BEEN IMPLEMENTED
* PROVIDE GUIDANCE ON ACCEPTABLE AGING MANAGEMENT PRACTICES.

SCOPE OF SRP-LR

* PROVIDES A CHECKLIST FOR DETERMINING THE SUFFICIENCY OF AN APPLICATION.

* REVIEW DEFINED BY 10 CFR PART 54 AND LIMITED TO:

  - SSCs IMPORTANT TO LICENSE RENEWAL

  - AGE-RELATED DEGRADATION CONCERNS

* CONCERNS ARISING FROM CLB ISSUES ARE OUTSIDE THE SCOPE OF SRP-LR. CHANGES TO THE CLB WILL BE REVIEWED IN ACCORDANCE WITH THE GUIDANCE PROVIDED BY NUREG 0800.

* "LIVING DOCUMENT" WHICH WILL BE REVISED AS EXPERIENCE IS GAINED FROM INDUSTRY TECHNICAL REPORTS, PILOT PLANT APPLICATION REVIEWS, AND ONGOING RESEARCH AND AS A RESULT OF PUBLIC COMMENTS.

ORGANIZATION OF SRP-LR

* Development of SRP-LR

* Three major sections:

  - Part A - General Information and Discussion

  - Part B - Systems

  - Part C - Generic Components and Structures

* General Structure for SRP-LR Part B and C Sections

  - Review Responsibilities
  - Areas of Review
  - Acceptance Criteria
  - Review Procedures
  - Findings
  - Implementation
  - General Information
  - References

SRP-LR PART A - GENERAL INFORMATION AND DISCUSSION

* DESCRIBES THE PURPOSE, SCOPE, AND ORGANIZATION OF SRP-LR.
* DESCRIBES THE GENERAL REQUIREMENTS OF THE LICENSE RENEWAL RULE.
* PROVIDES A DETAILED CHECKLIST TO BE USED WHEN EVALUATING THE SUFFICIENCY OF A LICENSE RENEWAL APPLICATION.

APPENDIX A

* PROVIDES GUIDANCE FOR THE STAFF REVIEW OF THE APPLICANT'S SCREENING METHODOLOGY FOR IDENTIFYING SSCs IMPORTANT TO LICENSE RENEWAL.

SRP-LR Part B - Systems

* Provides guidance for the staff system level review to determine if renewal applicants have:

  - Identified aging mechanisms for SCs of concern and

  - Described established effective programs, program modifications, or new programs which address aging degradation concerns or

  - Provided analysis of age-related degradation which establish that degradation for the renewal term is not significant.

SRP-LR Part B (cont.)

* Organized on a system basis
  - Not all systems expected in a renewal application are specifically included in SRP-LR Part B.
  - A generic system chapter provides staff guidance for systems not specifically addressed.
* For individual components or structures within a given system, the appropriate sections of SRP-LR Part C are referred.

SRP-LR PART C - GENERIC COMPONENTS AND STRUCTURES

* PROVIDES REVIEW CRITERIA FOR SPECIFIC GROUPS OF COMPONENTS AND STRUCTURES.

* SRP-LR PART C EXPECTED TO BE THE DOMINATE PART OF SRP-LR FROM A TECHNICAL VIEW POINT.

Review Criteria

* SRP-LR contains specific criteria related to managing aging degradation concerns for individual SSCs.

* In general, these new criteria:

  - Are additional inspections or analysis which may or may not be currently required but which will be used to determine the actual status of SCs from an aging perspective.

  - Are derived from the NPAR Program, plant experience, and engineering judgement.

* These criteria will evolve as a result of public comments, Industry Technical Reports, and pilot plant reviews.

EXAMPLES OF SPECIFIC NEW REVIEW CRITERIA

* SRP-LR C.1.1 PIPING

- THE LICENSEE SHALL HAVE A PROGRAM FOR MEASUREMENT OF PIPE WALL THINNING, PARTICULARLY FOR PIPING EXEMPT FROM ASME CODE SECTION XI BUT IMPORTANT TO LICENSE RENEWAL.

- THE LICENSEE SHALL VERIFY USING PLANT-SPECIFIC FATIGUE ANALYSIS THAT THE ASME SECTION III CUMULATIVE USAGE FACTOR ALLOWABLE OF 1.0 WILL NOT BE EXCEEDED. CONSIDERABLE FATIGUE GUIDANCE IS PROVIDED IN THE REVIEW PROCEDURE SECTION.

- THE LICENSEE SHALL HAVE A PROGRAM TO SAMPLE FOR STRESS CORROSION CRACKING.

- THE LICENSEE SHOULD INVESTIGATE POTENTIAL FLOW REDUCTION PROBLEMS.

Examples of Specific New Review Criteria (cont.)

* SRP-LR B.4.4 Emergency Diesel Generators (EDGs)

- EDG goal reliability has been met for last 10 years and all operating boundaries are currently within acceptable limits established by the manufacturer.

- Engine crankshaft and generator alignment is within the manufacturer's recommendations.

- Main bearing wear should not exceed the manufacturer's recommendation.

- Fatigue cracking of connecting rod bearings should not exist.

- No gear fatigue or excessive wear should be found.

- Turbochargers should be free from signs of ingestion damage, fatigue cracking, and bearing damage.

Implementation of SRP-LR

* License renewal application received.

* Application sufficient to commence detailed review.

* Review of screening methodology.

* Review from a systems, component, and structure perspective.

* Integration into a composite safety evaluation report.

FUTURE REVISIONS

* FUTURE REVISIONS WILL BE BASED UPON:

  - PUBLIC COMMENTS.

  - EXPERIENCED GAINED FROM THE REVIEW OF INDUSTRY TECHNICAL REPORTS.

  - EXPERIENCED GAINED FROM THE REVIEW OF THE PILOT PLANTS.

  - EXPERIENCED GAINED FROM THE NPAR PROGRAM.

SAFETY GOALS - BACKUP SLIDE

* IMPLEMENTATION OF DG-1009 AND SRP-LR WOULD NOT RESULT IN ADDITIONAL RISK TO LIFE OR HEALTH DURING THE RENEWAL TERM.

  - THE FOCUS IS ON ASSURING OPERATION OF SSC WHICH ARE IMPORTANT TO LICENSE RENEWAL AND ARE SUBJECT TO AGE-RELATED DEGRADATION.

  - DESIGN CHANGES WOULD ONLY OCCUR WHEN SYSTEMS OR STRUCTURES ARE JUDGED NOT ACCEPTABLE FOR CONTINUED OPERATION DURING THE RENEWAL TERM.

* IMPLEMENTATION OF DG-1009 AND SRP-LR WOULD NOT INCREASE SOCIAL RISKS TO LIFE AND HEALTH ABOVE THOSE CALCULATED FOR PRESENT PLANT OPERATION.

EXAMPLES OF NEW REVIEW CRITERIA

* SRP-LR C.1.3 PUMPS

- THE LICENSEE SHOULD HAVE A PROGRAM TO DETECT SMALL FLAWS CAUSED BY THERMAL EMBRITTLEMENT AND STUD CORROSION.

- THE LICENSEE SHOULD CONDUCT BOTH SURFACE AND VOLUMETRIC INSPECTIONS OF PUMP BODIES.

* SRP-LR C.1.4 HEAT EXCHANGERS

- THE LICENSEE SHOULD EVALUATE THE HEAT EXCHANGERS FOR MINIMUM WALL THICKNESS AND CONDUCT APPROPRIATE SAMPLING.

* SRP-LR CIVIL STRUCTURES

- MANY ONE-TIME ONLY INSPECTIONS ARE REQUESTED OF STRUCTURES AND FOUNDATIONS TO ESTABLISH CURRENT CONDITIONS.

Examples of New Review Criteria

* SRP-LR B.4.4 Emergency Diesel Generators

- This chapter contains six one-time tests and engine condition reviews.

* SRP-LR B.3.1 Primary Containment System

- Licensees should commit to RG 1.35 (ISI of ungrouted tendons in prestressed concrete containment structures) and RG 1.90 (ISI of prestressed concrete containment structures with grouted tendons).