ENCLOSURE

SAFETY ASSESSMENT

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

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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 valve (MOV) capabilities. Data 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 Cleanup (RWCU) suction lines were requested. This request was the result of a BWROG and NRC May 24, 1990 meeting which concerned the applicability of the Idaho National Engineering Laboratory (INEL) test data performed to resolve Generic Issue 87. The NRC interpretation of the data is provided 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 valve 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). In response to this NRC concern, the BWROG developed a "generic" safety assessment to document the adequate safety margin of BWR plants, and provided this assessment to the NRC (Reference 1). The BWROG assessment 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.

On October 25, 1990, the NRC issued Generic Letter (GL) 89-10, Supplement 3 (Reference 2) requesting licensees to develop a plant-specific safety assessment. This document fulfills the NRC request and is based on: 1) the Reference 1 BWROG Safety Assessment, 2) the GE evaluation of the applicability of the BWROG report to Plant Hatch (Reference 3), and 3) SCS and Bechtel assessments of environmental qualification (EQ) and flooding considerations (References 4, 5, and 6).

2.0 SUMMARY

This plant-specific safety assessment shows that the failure of the HPCI, RCIC, or RWCU isolation valves to promptly isolate following a postulated design-basis guillotine line break is not a significant safety concern.

The isolation MOVs of concern were selected, sized, and set using good engineering judgment based on the state of the art at the time of purchase. Plant Hatch design provides for early means of leak detection before a complete design basis-pipe failure could occur. Materials were selected for low probability of pipe failure. Inservice inspection and testing (ISI) 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 statically tested based on IE Bulletin 85-03. Emergency Operating Procedures (EOPs) provide a means of rapidly reducing the MOV service conditions if a pipe break occurs.

This assessment, employing a realistic integrated systems approach, concludes the existing affected MOVs for the HPCI, RCIC and RWCU Systems 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 mimicking the likely demand event, a pipe leak. System isolation will occur before the postulated design-basis event high flow/high differential pressure condition. Based on this, the presently installed equipment does not represent an undue risk to the health and safety of the public.

The Plant Hatch Final Safety Analysis Reports (FSARs) 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 (LBB) concept as a basis for pipe restraint removal in light water reactors.

These environmentally qualified MOVs, which perform the isolation function, have shown adequate operability for many years during normal 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.

Leak detection equipment exists in both Plant Hatch units to detect a small pipe leak condition and initiate system isolation. Small leaks represent such a small quantity of fluid flow escaping from the system that normal flow parameters will not be noticeably changed. Features to detect a small leak are summarized in References 7 and 8, and are sensitive to leakage flow rates less than 25 gpm.

MOV isolation performance will be the same as already demonstrated by multiple isolations (both during periodic testing and inadvertent initiations) of these valves during plant operation.

A realistic assessment of the consequences of a postulated design-basis pipe break condition, leads to the conclusion there is adequate safety margin to protect the reactor core and isolate the system successfully. The analysis in Reference 9 shows that RCIC or HPCI alone is capable of providing adequate core cooling once a broken high-energy line is isolated. Plant Hatch is licensed with the SAFER/GESTR-LOCA methodology (Reference 11). Application of realistic analyses has shown that only one of the Emergency Core Cooling System (ECCS) pumps is adequate to provide core cooling. This capability is the bases for the EOPs for Plant Hatch.

Additionally, the HPCI, RCIC, and RWCU lines are equipped with two isolation valves. If either of these valves closes, isolation is accomplished. Any action that 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 Plant Hatch EOPs.

Georgia Power Company (GPC) has also assessed the impact of an extended high-energy blowdown outside containment relative to reactor building EQ and flooding. These assessments have concluded an extended blowdown would not have an adverse impact on the EQ of any component located in the affected areas, and equipment submergence due to potential flooding in the Reactor Building will not affect the ability to safely shut down the reactor.

It is not expected HPCI/RCIC/RWCU System isolation MOVs will be challenged at high-flow design-basis accident (DBA) conditions because of LBB 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 is 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 ECCSs are designed for a much larger break than these small line ruptures. 10 CFR Part 100 offsite dose limits are not expected to be exceeded even with a delayed isolation response for any of these three systems.

It is recognized that INEL testing has identified anomalous valve behavior in the test valves under their test conditions. The BWROG and GPC are following this testing and reviewing engineering data as it becomes available for plant application. Based on the data applicability, GPC personnel are reviewing safety-related MOVs to assure the valves will operate on demand under design-basis conditions.

GPC actions initiated in response to GL 89-10 are proceeding with consideration of the INEL data to prioritize valves for review and testing.

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). Both units at Plant Hatch have multiple channel, redundant leak detection monitoring of the high-energy system lines external to the containment (Reference 7 and 8). This monitoring is sensitive to small leaks and causes both an alarm in the control room and automatic isolation signals to the leaking system's isolation MOVs. Isolation signals are designed to initiate MOV closure 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 FSARs, includes the evaluation of a loss-of-coolant accident (LOCA) 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 crack initiation. Environmental effects are not considered significant. Piping materials used at Plant Hatch are either Type 304 stainless steel or A-106 Grade B carbon steel, both of which have adequate toughness to qualify for the LBB justification. 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 would 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 the Plant Hatch FSARs.

In general terms, the LBB concept is based on the fact 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 having the ability to monitor the leak rate from any postulated 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 (Reference 9), 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 (GDC) 4, 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, which 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 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

Table 1 lists the values of parameters used in the critical crack length and leak rate calculations for the Reference 1 safety assessment. 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 4-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 10.

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 steam flow restriction (crack). Thus complete MOV closure will occur under these conditions. The RWCU cold-water 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.

The piping materials used in Plant Hatch were evaluated in Reference 3 using the same concepts and methodology as described in the Reference 1 BWROG Safety Assessment. Based on the results of LBB investigation and the leak detection monitoring and isolation systems, as discussed in subsection 3.4, it is concluded that piping systems in question (HPCI and RCIC steam supply lines, and the RWCU water supply line), are expected to develop a detectable leak long before the point of incipient rupture.

The pipe diameters covered in the BWR Owners' Safety Assessment range from 4 inches to 16 inches. The inlet of the RWCU for Plant Hatch is a 6-inch line from the reactor recirculation system. There are two 6-inch RWCU isolation valves: one inboard and one outboard of Primary Containment. After the 6-inch outboard isolation valve, the RWCU line branches into two parallel 3-inch lines. Each of these 3-inch lines is connected to a RWCU pump. The pump discharge lines are connected to a 4-inch line for the remainder of the RWCU piping. Since the small leakage within the 3-inch line may exceed a critical crack length for early leak detection, the effect of a complete pipe break at the 3-inch line was evaluated.

Since the 3-inch line is smaller than the 6-inch isolation valve, the break flow will be limited by the size of the 3-inch piping (i.e., the flow will choke first at the broken pipe). Consequently, when the 6-inch isolation valves receive an isolation signal and start to close, the differential pressure across the isolation valves will be significantly less than for a 6-inch break. As the isolation valves continue to close, the break flow will eventually choke at the valve. It is estimated that choking at the valve will occur at approximately 66 percent of the valve stroke. This is a conservative estimate because the 3-inch line also provides a higher resistance, thus the limiting area of the valve will be smaller. Since the choked flow condition (high differential pressure condition) occurs near the end of the valve stroke, the valve will continue to close as discussed in Section 3.7. Furthermore, since the two isolation valves do not necessarily close at the same rate, the stalling of one of the valves near the end of its stroke will allow the other valve to close further, which will relieve the high differential pressure across the stalled valve.

Even if it were postulated that the valve could not close, the reactor inventory loss through the 3-inch line can be compensated by any one of the low-pressure ECCS pumps as demonstrated in Reference 11.

The analysis performed for the Reference 1 BWROG's Safety Assessment assumed that all the pipe thicknesses were Schedule 80. All the Hatch 1 piping in the systems of interest is Schedule 80. However, in Hatch 2, the 10-inch HPCI line is Schedule 100, and the 4-inch line is Schedule 120. To assess the significance of this difference in wall thickness, an evaluation of the leak rate at the critical crack length for the 4-inch Schedule 120 pipe was conducted. This evaluation showed that the critical crack length for a 4-inch Schedule 120 pipe will be about 6 percent lower than that for a 4-inch Schedule 80 pipe. This difference is considered insignificant. Therefore, the conclusion is that the generic LBB assessment is applicable to Plant Hatch when considering the plant's unique piping configuration.

3.4 Leak Detection Monitoring and Isolation

Both units at Plant Hatch have been designed for compliance to GDC 54 of the Code of Federal Regulations, Part 50, Appendix A.

> "Piping systems 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 Plant Hatch, 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 isolate that system when a leak condition is detected. The sensors and logic are supplied 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 details of the Plant Hatch temperature monitoring arrangement and design (both ambient and differential) can be found in Reference 12.

The range of plant system areas protected by leakage instrumentation 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 the Plant Hatch Technical Specifications. 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 also reach the temperature limit and allows detection of smaller cracks.

In addition to temperature monitoring in the RWCU System, Plant Hatch has cold-water low-flow leakage monitoring capability. This cold-water, small-break, redundant, differential flow monitoring leak detection capability measures flow into and out of the system. It has an isolation limit of less than 100-gpm flow mismatch between the system inlet and outlet. 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 startup 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. Radiation monitoring and sump monitoring systems which can be used to detect small leakage in certain areas are described in References 12, 13, and 14.

Operating experience has shown relatively quick operator response to potential leakage 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 Plant Hatch can detect the small leakage condition and initiate isolation long before a pipe break condition would develop. Therefore, the combination of the LBB approach, in conjunction with the leak detection capability provides early isolation at less than design-basis conditions for a potential pipe break that challenge the MOV 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 10 CFR Part 100 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 lb total, of which 120,000 lb 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 10 CFR 10_ limit.

3.6 Equipment Qualification

Equipment qualification of the affected MOVs has been performed to pipe break harsh environment envelope bounding conditions, which are much worse than small leak environmental conditions. For example, certain area temperature monitor leak detection isolation setpoints for HPCI and RCIC are approximately 170°F, which is far below the equipment qualification temperature profiles for those areas following a postulated line break. Satisfaction of equipment qualification requirements assures continued equipment safety function performance including MOVs up to the bounding equipment qualification conditions. Therefore, no equipment qualification 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 LBB 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, including Plant Hatch. Some of these isolations have occurred at or near 100-percent 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 near 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 subject the valves to high differential pressure 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.

3.8 Piping Inspections

The HPCI steam supply line is a carbon steel Class 2 line and receives periodic inspections per the requirements of 10 CFR 50.55a and ASME Section XI Inservice Inspection (ISI), 1980 edition with addenda through Winter 1981. This includes a sample of volumetric and surface exams on weldments, and a hydrostatic test each 10-year interval. This piping also receives operational pressure leakage tests each 40-month ISI period. Both the HPCI and RCIC Systems are functionally tested at least once per three months in accordance with Technical Specifications and ASME Section XI Inservice Testing (IST) requirements. The exams on the subject HPCI piping have shown no propensity for service-induced flaws. Erosion/corrosion programs do not currently include the HPCI or RCIC piping because of the small amount of time the systems actually operate.

The RWCU water supply line outside containment is constructed of stainless steel, and is considered susceptible to intergranular stress corrosion cracking (IGSCC) per NUREG-0313, Revision 2. Georgia Power Company currently performs augmented examinations on these welds per our commitments to Generic Letter 88-01 (Reference 15). These commitments, documented in References 16 and 17, require examination of about 10% of the RWCU welds outside containment (approximately 7 welds) each outage. To date GPC has examined seven welds in each unit's nonsafety-related RWCU piping and detected no reportable IGSCC crack-like indications.

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 has been evaluated from a realistic perspective, as was the postulated small leak condition. A realistic review, without all of the design basis assumptions, was conducted because of the low probability ($4x10^{-4}$/year) of a high-energy line break in one of these systems. Any MOVs at Plant Hatch 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-percent 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 mounted in the supply lines very close to one another, separated only by the primary containment wall. Upon receipt of isolation signals they will not close at exactly the same time. This is because of small physical differences, as well as the fact the outboard isolation valves are driven by AC motors while the inboard valves 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.

A full HPCI steam line break will reduce 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. A HPCI, RCIC, or RWCU line break will result in the entry into secondary containment (SC) control portion of the Plant Hatch EOPs. Inability to isolate the break flow would likely lead to exceeding the maximum safe operating values of SC/Temperature (SC/T), SC/Radiation (SC/R), or SC/Area Water Level (SC/L) in the room where the break occurs. This will result in an operator instruction to scram and begin reactor cooldown. If the situation degrades and spreads so that equipment in more than one area (or room) is threatened, an instruction to rapidly depressurize the reactor is provided which will quickly reduce the break flow and mitigate the event. A detailed discussion of the operator actions for an unisolatable line break in secondary containment is contained in Reference 18. 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 size 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 primary containment in one of the three systems, the impact on the nuclear system would be less severe than a DBA. The high-energy lines are small lines (compared to the DBA) and would require less 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 as would HPCI or RCIC alone. Existing FSAR 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). Section 4.4 details the safety assessments performed to address EQ and flooding assuming a DEGB occurs and the HPCI, RCIC, or RWCU isolation valves do not immediately isolate the line break. The assessments showed that adequate safety systems will remain available to allow the plant to be brought to and maintained in a cold shutdown condition.

Therefore, Plant Hatch has 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 lines 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 10 CFR Part 100. 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 10 CFR 100. Therefore, any slightly longer valve stroke time for the RWCU isolation valves will not result in noncompliance with the requirements of 10 CFR Part 100.

4.4 Environmental Qualification (EQ)

EQ for Reactor Building components utilizes the enveloped response (e.g., temperature profiles) for all postulated line breaks that affect that particular area. For example, the Unit 1 Reactor Building 130' elevation temperature profile, shown in Figure 1, is controlled by the main steam line break outside containment from 0 to 6 seconds, and the HPCI line break after that. At 63 seconds, the HPCI line is isolated and the temperature on the 130 elevation gradually falls. The components in the Reactor Building are qualified to temperatures at least as severe as the Figure 1 enveloping temperature profile. For Plant Hatch, the RCIC and RWCU line breaks are not limiting for the areas of interest, and the HPCI line break (which has break mass flux more than 10 times RCIC for a guillotine break) dominates.

To assess the impact on EQ for this issue, the qualification data packages for all safety-related equipment in the Reactor Building were reviewed assuming a full HPCI line break for 10 minutes (rather than the 63 seconds currently assumed). A full HPCI line break results in a 100-percent humidity condition, even for conditions of prompt isolation. This is a very conservative assumption, because it assumes a guillotine line break, conservative FSAR break flow rates, failure of one isolation valve, and no movement of the other valve for 10 minutes. It also takes no credit for the probable reactor scram on water level or Group I isolation, or the reactor depressurization. It also takes no credit for partial valve closure which would reduce the mass flux out the break.

Even with the longer isolation time, the peak temperature will not increase significantly, since the Reactor Building is large and equipped with blowout panels. Therefore, essentially atmospheric pressure exists in the building and peak temperature remains at the saturation temperature at that pressure (approximately 212°F), regardless of the isolation time. Time duration at this peak pressure increases without prompt isolation, but is still bounded by the EQ test profiles.

Based on GPC's review of component qualification test profiles and material analysis, a 10-minute HPCI blowdown would not have an adverse impact on the EQ of any component located in the reactor building. This review was conducted on all safety-related equipment in the Reactor Building, which included such items as valve operators, motor control centers, cabling, transmitters, terminal blocks, cooling fan motors, and ECCS pump motors.

4.5 Flooding

An assessment of the flooding implications of this issue was also made for Plant Hatch Units 1 and 2. Walkdowns were performed in the Reactor Building to identify existing flood control measures (e.g., curbs, drains) and probable water flow paths. This information was used to assess the impact of spraying and submergence of safety-related equipment.

For Plant Hatch, the RWCU line break outside containment is most limiting. A guillotine line break of the HPCI line without prompt isolation would release a large amount of steam, which condenses. However, virtually all the water will end up in the torus room (elevation 87') or the HPCI Room at Plant Hatch. Flooding of the torus room is not a serious concern, as the floor area is large enough to keep the flood level low. Therefore, ECCS and containment cooling systems would be expected to remain operable.

The RWCU guillotine line break without prompt isolation can release a large amount of water on to the 158' elevation of the Reactor Building (in excess of the drainage system capability). Much of this water could flow down to the 130' elevation, and end up in the ECCS diagonal rooms. Plant walkdowns included measurements of the equipment heights above the floor in the ECCS diagonals, and measurement of berm heights on the 158' and 130' Reactor Building elevations. The stairway and berm configurations of both units would tend to dispose most of the water to one ECCS diagonal, leaving the other relatively dry. (Sufficient equipment is available in either diagonal to safely shut down the reactor in this postulated RWCU line break condition.)

The amount of water assumed to flow out the break was estimated assuming. 1) conservative FSAR break flow methodology with no credit for flashing, 2) simultaneous loss-of-offsite power with complete failure of one RWCU isolation valve, and 3) partial valve closure (50 percent) of the other isolation valve. The line was assumed isolated after 10 minutes. Even with these conservative assumptions, sufficient equipment will remain operable to safely shut down the reactor.

5.0 CONCLUSIONS

Because of the LBB 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 ECCSs 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 10 CFR Part 100 requirements.

TABLE 1

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

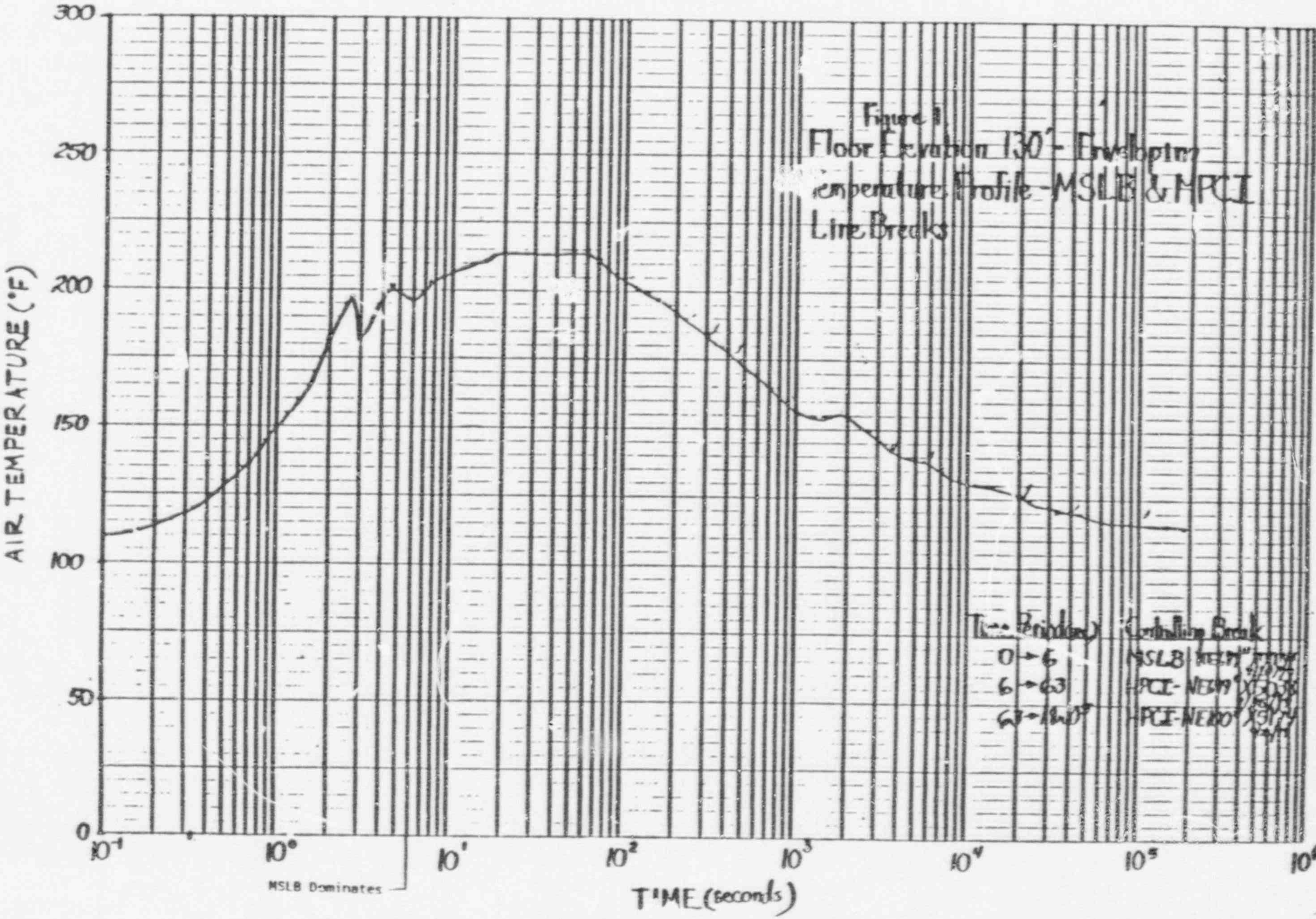
Pipe Thickness	:	Schedule 80
Pipe Internal Pressure	:	1050 psi
Temperature	:	528°F
Normal Operation Bending Stresses	:	4 ksi
Material	:	Stainless Steel or Carbon Steel

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

FIGURE 1 UNIT 1 - FLOOR ELEVATION 130 FT - ENVELOPING TEMPERATURE PROFILE - MSLB AND HPCI LINE BREAKS



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