GE-NE-A0005845-01

SAFETY ASSESSMENT

BWR EMERGENCY CORE COOLING SYSTEM SUCTION STRAINERS

GE Nuclear Energy
For
BWR Owners' Group

March 1994

R. C. Mitchell

Approved by: S. J. Stark

S.J. Stark, Manager
BWR Owners' Group Projects

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

The only undertakings of General Electric Company respecting information in this document are contained in the contract between the Boiling Water Reactor Owners' Group and General Electric Company (i.e., the Standing Purchase Orders for the participating utilities in effect at the time this report is issued) and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than that for which it is intended, is not authorized; and with respect to any unauthorized use, General Electric Company makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

TABLE OF CONTENTS

SAFETY ASSESSMENT FOR BWR EMERGENCY CORE COOLING SYSTEM SUCTION STRAINERS

1.0 Introduction

Unresolved Safety Issue (USI) A-43: Containment Emergency Sump Performance was resolved by the Nuclear Regulatory Commission (NRC) in 1985. This issue dealt with the availability of adequate cooling water following a Loss of Coolant Accident (LOCA) when long-term recirculation of cooling water through the Residual Heat Removal (RHR) suction strainers must be initiated and maintained to prevent core-damage. The closure of this issue concentrated on the Pressurized Water Reactor (PWR) sump debris and screens with secondary consideration of Boiling Water Reactor (BWR) suppression pools. Regulatory Guide 1.82, Revision 1 [Reference 1] was issued in November 1985 to address regulatory guidance for long term recirculation cooling following a LOCA. The Regulatory Guide has forward fit guidance for new applications. Recent events at BWRs have caused the NRC to reconsider USI A-43.

On July 28,1992 a relief valve discharge occurred at Barsebäck 2, a Swedish BWR designed by Asea-Atom. Unlike U.S. BWRs, the electrically operated relief valves at Barsebäck are piped to discharge steam directly into the drywell airspace. The discharge resulted in a steam jet that caused damage to some piping insulation in the drywell. Subsequent carryover of the insulation debris to the suppression pool resulted in blockage of the containment spray suction strainers in about one hour. Prior Swedish utility calculations had indicated that blockage should not have occurred for at least ten hours. This event brought into question the previous understanding of modeling of debris generation, debris transportation to the wetwell, debris transportation within the wetwell, and the head loss characteristics of debris blockage of the suction strainer.

Subsequent to the Barsebäck event, the Perry plant experienced fouling and mechanical deformation incidents of RHR suction strainers in 1992 and early 1993. An LER [2] was issued on the incidents. These occurred due to debris that was present in the suppression pool. The resulting blockage was not associated with a transient or an accident event involving steam release in the drywell. The significance of these incidents was that they developed the concern for the combined effects of insulation and particulate matter (i.e., post LOCA generated debris and corrosion products). When carried to the suppression pool, the insulation acts as a filter and ECCS suction strainer blockage occurs due to the filtering of the other debris.

On May 11, 1993, the NRC issued Bulletin 93-02: DEBRIS PLUGGING OF EMERGENCY CORE COOLING SUCTION STRAINERS [3]. This dealt with the issue on potential sources of non-insulation fibrous material in the containment and the need to remove it.

On July 2, 1993, Pennsylvania Power & Light submitted a voluntary Licensee Event Report (LER) to the NRC [4]. This dealt with the combined effects of fibrous insulation, coating material, and corrosion products on suppression pool suction strainer blockage during a postulated LOCA event at Susquehanna 1. This LER further pointed out the potential for combined debris effects.

Because of this recent experience, the NRC has embarked on a program to evaluate loss of ECCS due to debris blockage of suction strainers. Science & Engineering Associates has been contracted by the NRC to assist in this effort. Work will focus on performing an analysis for the BWRs to evaluate debris transport models, estimate core damage frequency (CDF) impacts, and identify mitigating actions.

On February 18, 1994, the NRC issued Supplement 1 to Bulletin 93-02. This requested that all BWRs respond with compensatory actions to mitigate the potential for loss of Emergency Core Cooling System (ECCS) capability due to blockage of the suctions strainers.

The purpose of this safety assessment is to provide a generic perspective on BWR plant safety relative to ECCS debris and strainer blockage. It addresses the safety significance of this issue and provides a basis for demonstrating continued plant safety pending final resolution of the issue. BWR Owners' Group (BWROG) members may utilize or reference this assessment in their response to the Bulletin supplement. The assessment will cover the following:

1. The potential for a Barsebäck type event.

2. Strainer blockage from foreign material.

3. Probability of strainer blockage events and impact on plant safety.

4. The availability of cooling from alternate water sources.

Concurrently, the BWROG is examining this concern in more detail in order to:

1. Develop plant specific evaluation methodologies that will allow each member to assess the need for design or operational changes, and

2. Develop potential design or operational changes that can be used to resolve the issue where required.

2.0 Background

The design basis for BWRs (late BWR/4s, all BWR/5s and BWR/6s) is that each ECCS pumping loop is provided with at least one separate suction strainer. These suction strainers are designed such that adequate Net Positive Suction Head (NPSH) is provided to the ECCS pumps assuming that the suction strainer flow area is 50% plugged and that for BWR/6 plants the strainers should not become more than 50% plugged following 100 days of post LOCA operation [5].

Earlier BWRs utilized a "ring-header" suction design in which all of the ECCS pumps were provided suction flow from a common circular header pipe. The header pipe was provided with flow from the suppression pool by three or four pipes connected to the suppression pool torus at approximately evenly spaced distances around the torus lower circumference. Each suction pipe from the suppression pool was provided with a suction strainer. The ECCS pumps receive water from the ring header. The ring header and the piping connections to the suppression pool, along with their associated suction strainers, were designed such that adequate NPSH was available to all of the ECCS pumps with any single suction strainer 100% blocked.

There are multiple and redundant safety systems available for emergency core cooling in U.S. BWRs. In combination with the allowance for suction strainer blockage in earlier and more recent designs, these systems have been considered a justifiable and prudent design basis for ECCS.

In closing USI A-43 the NRC developed regulatory guidance with regard to debris and strainer design. Regulatory Guide 1.82, Rev. 1 was issued to provide forward fit guidance in assessing the potential for insulation to be dislodged in a LOCA. It acknowledges the complexity of estimating insulation debris generated by pipe break jet forces and the modeling of the predicted jet envelope that determines the zone of influence. The guidance and the specified zone of influence are determined from a review of the large break Design Basis Accident (DBA). The DBA represents an extremely conservative scenario that postulates an instantaneous, double ended rupture of the recirculation pipe with free flow out of each end of the pipe. In the case where the steam or feedwater line rupture would produce more insulation debris, the same type of rupture is imposed. The Guide further states that the type of insulation utilized must be individually addressed. Different types of insulation would require consideration of such factors as the insulation material itself, whether it is encapsulated, and how it is fastened to the pipes.

There are major differences between the Barsebäck plant and the plant designs employed by U.S. BWRs. Some of the differences include the containment layout, safety/relief valve discharge piping, automatic containment spray initiation at Barsebäck versus manual initiation for U.S. BWRs, types of insulation, insulation encapsulation, and alternate sources of water. These are discussed in the following assessment.

3.0 Summary

This assessment concludes that the Barsebäck event cannot occur in U.S. BWRs because they all have the relief valves piped directly to the suppression pool. If a LOCA event were to occur at U.S. BWRs, there are significant differences in the plant design when compared to Barsebäck that would help to mitigate the issue of ECCS suction strainer plugging. Additionally, all U.S. utilities have responded to the NRC Bulletin 93-02 that addressed transient fibrous materials and their removal from the containment.

Large pipe ruptures (DBAs) are not expected and any pipe failure will be preceded by detectable leakage. Current plant designs, operating procedures, and Technical Specifications are adequate to assure detection and correction of leakage in the primary system pressure boundary piping. This detection and correction are designed to occur well before any challenge occurs to the plant ECCS system performance due to suction strainer blockage.

U.S. BWRs have symptom based Emergency Operating Procedures (EOPs). These procedures lead the operators through responses to any decline in reactor water level. These include the utilization of any available alternate water sources that can be employed to inject water into the reactor vessel. These actions would be called for without the need for recognition of suction strainer blockage.

In the regulatory consideration of the design basis LOCA, a plant specific evaluation may not result in any significant strainer blockage (e.g., metallic insulation). Each plant needs to evaluate their specific insulation design, piping, and containment to confirm their ability to meet their licensing basis ECCS requirements. The BWR Owners' Group is developing methodologies to assist in these individual plant assessments and potential design and operational options, should they be required. If the individual plant assessment for the assumed LOCA shows a significant potential for ECCS suction strainer blockage, this safety assessment concludes that continued operation is acceptable while long term actions are determined and implemented. This conclusion is based upon the low probability of the initiating event, leak-before-break, and operator recovery actions contained in the plant EOPs.

4.0 Safety Assessment

4.1 Barsebäck Evaluation

4.1.1 Barsebäck Event

The discharge of the relief valve into the containment at Barsebäck was the result of a maintenance error. The electrically operated solenoid was reinstalled in a condition that caused it to be open. As the reactor was being brought up in pressure, the rupture disk on the relief valve discharge released as designed. The resultant steam blowdown destroyed some of the mineral wool insulation which went to the suppression pool. This was further aggravated by the automatic initiation of the containment spray pumps which washed more insulation to the suppression pool. The ECCS pumps were turned off, but the containment spray continued to run. The significance of the event, as stated in Section 1.0, was that the containment spray suction strainer blocked much quicker than had been previously calculated to occur. This raised the concern for the potential blockage of ECCS suction strainers in a design basis LOCA.

The Barsebäck containment spray suction strainer blockage incident had several factors that compounded the incident that would not be typical for any U.S. BWR [6]. These factors include the pipe discharge into the drywell, insulation type, insulation jacketing, containment layout, automatic initiation of the containment spray, and pool mixing. The differences are as follows:

- The Barsebäck relief valve discharged directly into the drywell where it impinged upon the insulation. In U.S. BWRs, all of the electrically operated relief valves are piped to the suppression pool and discharge under water. There is a limited number (12 units with Mark I containments) of the earlier U.S. BWR plants that have spring operated safety valves which discharge into the drywell. Maintenance of these valves is only done after removal from the plant and they are retested to assure proper lift pressure before being reinstalled in the steam line. This precludes the possibility of the Barsebäck occurrence in U.S. BWRs. Further, these safety valves would only open if the lower set point relief valves did not actuate or the pressure increase continued to the safety valve setpoint after relief valve actuation. To date, there has not been any discharge of BWR safety valves into the drywell from steam pressurization events.

There was one event of a lift of a spring operated safety valve that discharged into the drywell. This event was initiated by an overfill of the reactor vessel following a reactor scram at full reactor pressure. The vessel overfilled when the operator took manual control of the feedwater system. One unpiped safety valve lifted due to a water-hammer induced pressure spike in one of the steam lines. This one safety valve discharged through a rams-head discharge splitter onto two adjacent safety valve manual lifting handles. The handles wedged against the valve bodies with the valves in the partially opened position. These two safety valves depressurized the reactor into the drywell. This event should not occur in the future because high level (Level 8) feedwater pump trips have been installed, or are about to be installed, at all operating BWRs. Further, it was recommended that the safety valve lifting handles be removed. The handles were subsequently removed. These actions should prevent reoccurrence. It is noteworthy that no problem with insulation generated debris was reported for this event.

4.1.2 Barsebäck Event and LOCA Considerations

The Barsebäck event raised the concern of the impact of insulation generated debris in a LOCA event. There are differences in the Barsebäck design and that of U.S. BWRs which would help to mitigate the concern for strainer blockage. These include:

- The Barsebäck insulation was aged mineral wool with very different properties than the insulation used in most U.S. BWRs. The mineral wool was very easily destroyed and turned into dust-like particles. These particles were blown around with the steam and caked out like mud on many of the drywell surfaces while the rest was carried over to the suppression pool with the steam. Mineral wool causes a greater head loss when caked on the strainer than fiberglass insulation. The zone of destruction for mineral wool is greater than would be expected for other types of insulation.

  The U.S. BWRs primarily use reflective metallic insulation (RMI) or fiberglass insulation such as Nukon™. Each of these materials has quite different characteristics than the mineral wool. They have a higher resistance to destruction and their transport characteristics are quite different.

- The mineral wool at Barsebäck was jacketed with 0.04" riveted aluminum jacketing. Nukon™ is jacketed with 22 gage (0.03") 304 stainless steel held by latches. Testing of the jacketed Nukon™ properties demonstrated a reduced volume of debris in the zone of destruction with a lesser amount of fine particles available for transport in a postulated pipe break [7]. Even unjacketed Nukon™ would be more resistant to debris generation from longer term steam erosion than unjacketed mineral wool.

- Of the containment types used for U.S. BWRs, the Barsebäck containment most closely approximates a Mark II containment. Insulation debris generated at Barsebäck had a fairly direct vertical drop path into the suppression pool.. Any insulation falling on the drywell floor would be susceptible to washing into the pool earlier in an accident because Barsebäck had automatic initiation of containment spray. Additionally, the 96 Barsebäck downcomers are flush with the drywell floor. Domestic Mark IIs with raised downcomers would take longer for the water level on the drywell floor to rise high enough that the insulation laying on the drywell floor would wash into the pool. Depending upon water flow velocity and insulation settling, it may be less likely to wash into the pool. The manual initiation of containment spray might further delay the transport of insulation to the pool. The Mark I drywell design has a large hold-up volume in the drywell floor and a more tortuous path to the suppression pool. This includes jet deflector plates over the main vents to the torus. Consequently, insulation transport would be more difficult in a Mark I containment. Mark IIIs also have a large drywell floor hold-up volume and a less direct path from the drywell to the suppression pool. Each individual containment and containment type have unique features that would effect insulation debris transport.

- The Barsebäck suppression pool has a mixing propeller that enhances the debris suspension/transport in the pool and allows less debris to settle to the bottom of the pool. This mixing would carry more insulation to the ECCS suction strainers than would be expected without pool mixing. It is estimated that five percent of the insulation transported to the pool at Barsebäck made it to the suction strainers [6]. In a pool without additional mixing, the pool transport would be different and the time to blockage would be expected to be longer or possibly not occur. Some BWRs employ ECCS discharge jets to affect some pool water circulation in the pool cooling mode, but not the injection mode.

4.2 Strainer Blockage Considerations

4.2.1 Perry Strainer Blockage

The Perry plant incident demonstrated the combined effects of fibers and particulate debris on potential strainer blockage and the resultant importance of maintaining good housekeeping practices in the containment and suppression pool. There are unique features at Perry that are not present at all BWRs. These include:

- Perry is a Mark III containment that has an open suppression pool with personnel access above it (typical for all Mark III containments). Activities take place above the pool during refueling/maintenance outages and during plant operation. This results in a greater potential for direct introduction of foreign material to the suppression pool than is likely in a Mark I or II containment. Mark I and II containment personnel access is limited to plant shutdown periods.

- The Perry suppression pool was uniquely designed to have a high degree of mixing during residual heat removal pool cooling mode of operation to prevent thermal stratification. The resultant high pool velocities tend to keep debris suspended longer and thereby increase the likelihood of it being transported to the strainers. Other Mark III pools have less mixing and would not be as susceptible to strainer blockage in the pool cooling mode. Most importantly, in the initial phase of direct pool to vessel injection in response to LOCA conditions, the pool cooling mode would not be utilized.

4.2.2 Other Mitigating Considerations

Other factors can counteract the loss of NPSH due to strainer blockage in a postulated accident. These include the differences in strainer designs (shape, mounting, structural design, approach velocity) and alternate water sources available to the ECCS pumps for containment flooding.

Another factor that would help prevent loss of ECCS flow is the actual NPSH available versus that calculated to be available for NPSH design considerations. From a design basis accident evaluation perspective, Regulatory Guide 1.1 [8] requires use of the maximum calculated pool temperature with no credit for containment pressurization in determining the NPSH available for the pumps. Use of the realistically calculated pool temperature with time during the event would increase the available NPSH for the pumps, as would use of the containment pressure rise that occurs with the pool temperature increase. Attachment 1 notes show some realistic NPSH values for some plants.

4.2.3 Plant Specific Characteristics

Attachment 1 is a compilation of BWR Thermal Insulation Types and ECCS Characteristics for U.S. BWRs. This shows that almost all U.S. BWRs use Nukon™ and RMI that have significant differences from the mineral wool used at Barsebäck. Additionally, there are differences in the containment types, number of ECCS pumps, strainer arrangements and surface area, strainer approach velocities, pool mixing velocities, and available NPSH. These factors make each plant unique with regard to the potential for strainer blockage to occur and the impact of strainer blockage on event mitigative capability.

4.3 Containment Cleanliness

Information on pool cleanliness has been transmitted by the NRC to all licensees through Information Notice 93-34 [9] and Bulletin 93-02. The Bulletin required licensees to identify air filter or other temporary sources of fibrous material, not designed to withstand a LOCA, and take action to remove these from the containment. Plants have responded to the Bulletin and efforts are on-going to assure that potential sources of debris are not inadvertently introduced to the suppression pool.

The BWROG recognizes the need for suppression pool cleanliness and is evaluating the role of foreign materials in potential strainer blockage. Removal of foreign material will help to minimize the potential for ECCS suction strainer blockage. Some BWRs already have suppression pool cleaning systems that help remove foreign materials from the pool which lowers the potential for blockage of the strainers.

4.4 Alternate Water Injection

4.4.1 Alternate Water Sources

The overall design of the BWR plant includes several alternate sources of water that would be available to the operator for core cooling in the event that ECCS suction strainers should become plugged following a LOCA event. These include sources that can be pumped to the reactor with safety grade equipment and non-safety grade or non-Class 1E equipment. These potential success paths to core cooling are capable of taking suction from sources other than the suppression pool and are independent of the strainers.

In the case where off-site power is available or can be restored, the non-Class 1E condensate/condensate booster pumps can be used. These can pump the condenser hotwell water to the reactor to restore water level and cool the core for breaks above the top of the active fuel. This is the normal source of water available to the operator. Additionally, the Control Rod Drive (CRD) pumps would continue to supply CST water to the reactor as long as off-site or diesel power was available.

If ECCS injection flow rate were to drop off, the operator could switch to an alternate suction source. The high pressure core spray pumps (BWR/5&6) have the capability of being aligned back to the non-safety related Condensate Storage Tank (CST). BWR 2/3/4 plant designs also have this capability with the low pressure core spray pumps. As long as the ECCS pumps are still available, this potential source of water provides a substantial quantity of water for direct injection to the reactor.

Another source of water for injection to the reactor is the emergency RHR service water. This water source can be cross-tied to the low pressure coolant injection flow path to pump water from the cooling pond, lake, river, or other ultimate heat sink. This source of water is safety related and can on some plants be initiated from the main control room. The fire water pump is another potential way of getting water to the reactor using this same flow path. This pump, either diesel or normal bus driven, can be manually aligned to feed water into the reactor vessel.

While all of these systems may not be available for all BWRs or for all events, in the remote chance that all of the ECCS suction strainers were blocked, they do represent other means of delivering water to the reactor to prevent core damage under extremely improbable circumstances.

4.4.2 Operator Action

The BWR Owners' Group has developed an extensive set of symptom oriented Emergency Procedure Guidelines (EPGs). These have been utilized to develop plant specific Emergency Operating Procedures (EOPs) which uniquely focus the operator's attention on maintaining reactor water level under all circumstances. These naturally lead the operator to alternate sources of water for any declining water level event such as the blockage of the ECCS suction strainers. The BWROG has reviewed the adequacy of these EOPs and developed a guidance paper for operator review/awareness. This paper, entitled Operator Guidance for Potential Blockage of ECCS Pump Suction Strainers, has been distributed to all BWRs. A copy of the guidance paper is attached to this safety assessment. (See Attachment 2.)

As stated above, the operator is directed to take actions to overcome a potential strainer blockage event through these alternate water sources. Early recognition of degraded pump performance would allow the operator to take action. The key is that the operator responds to symptoms without the need to diagnose specific events while having sufficient time to respond. Specific recognition of strainer blockage is not required.

Strainer blockage should not occur instantaneously or simultaneously for pumps with individual strainers. The operator has instrumentation available to him (flow rate, discharge pressure, and declining reactor water level) to monitor performance of the ECCS pumps. Indication of degradation of the suction pressure should be available before failure of a pump. Further, because of the pool mixing and strainer locations throughout the pool, the pumps would not be expected to degrade uniformly or to fail simultaneously. If the operator's attention was diverted, the failure of the first pump should alert him to the potential failure of the other pumps. These subsequent failures could occur significantly later.

Reference 10 analyzed the time available to the operator to take corrective action should the strainers become blocked. Even with conservative assumptions that all strainers blocked instantaneously and simultaneously at 10 minutes into the recirculation line break accident (this compares to Barsebäck plugging at one hour with a material and a containment design which are more conducive to plugging), the calculated minimum time available for the operator to take corrective action was approximately 25 minutes. For failures that occur at sixty minutes into the accident, the calculated time available to the operator for action was 35 minutes. Other line breaks would result in longer times available for operator corrective action.

Revision 4 of the EPGs [11] has been implemented as EOPs by all BWR owners. The operators have been trained on the specific procedures developed from the guidelines. The procedures are based upon symptoms or operator recognition of the indications available. They do not require recognition of a specific event such as ECCS suction strainer blockage. As pumps potentially fail and the reactor water level decreases, the operator would be led through steps to restore the reactor water level through alternate water sources such as the RHR service water and to remove the heat from the containment.

If the event that caused strainer blockage were a break in a line above the top of the active fuel, it may be possible to establish long term cooling by using the shutdown cooling mode of the RHR system. This would provide for decay heat removal without concern for impending suction strainer blockage because the need for continuous injection of cooling water to the reactor would be eliminated. It would then be limited to make-up for the steam released through the unisolated leak until the shutdown cooling system had subcooled the reactor and terminated steaming. The EOPs would lead to that action for long term cooling.

With or without recognition of the strainer blockage, the operator has sufficient time to take action to protect the core. Reference 10 concluded that the operator should be successful 96 out of 100 times in preventing significant core damage. With recognition of the first pump's loss of NPSH/flow, the operator could take other actions to further increase the chances for success in this postulated accident.

4.5 Pipe Break Probability in BWRs

4.5.1 Probability of Rupture/Leakage

The probability of small pipe breaks (<2" in diameter) is generally recognized to be an order of magnitude greater than a large pipe break. The energy released by a small break would have a much more limited area of insulation damage. Additionally, the energy released to the suppression pool would most likely carry little or no insulation to the pool. Finally, the vent clearing action would be intermittent with much less energy and, thus, cause less pool mixing than would occur with a large break.

Large breaks are of interest because they would generate the greatest amount of insulation debris. For comparison, using the Regulatory Guide 1.82 assumption of total destruction over a length equal to three diameters of the failed pipe, a large pipe (24" diameter) would generate over fifty times more fine debris than a small pipe (2" diameter) for a postulated Double Ended Guillotine Breaks (DEGB). That conservatively assumes the same thickness of insulation on both pipes. Normally, the large pipe would have a greater thickness of insulation. Total insulation debris generated would be dependent upon the targets in the break vicinity.

Determination of definitive pipe break probabilities in BWRs is difficult because there is no experience base in BWRs of any pipe breaks, much less a DEGB. BWR recirculation systems are designed to the ASME code with appropriate safety margins applied. Small flaws can be detected by in-service inspection of piping before failures occur. Therefore, there is no failure history to calculate actual failure probabilities. Estimates must be made by analytical techniques or be based upon this operational experience without a failure history.

NUREG/CR-4792 provided analytical estimates of a DEGB probability for large pipes for a representative BWR [12]. These results indicated that, absent Intergranular Stress Corrosion Cracking (IGSCC) affects, the probability of a DEGB in the main steam, feedwater, or recirculation system ranged from 1.0E-12 to 3.82E-12 per reactor year.

Reference 13 provides an updated estimate for large breaks in the recirculation system based on current operational experience without a large break failure. This estimate is 7.51E-6 per reactor year. This is considered a conservative "upper bound" value. This estimate decreases with each additional year of reactor operating experience without such a failure.

NUREG/CR-4792 also provided analytical estimates of leakage. Absent IGSCC, the estimated probability of leakage ranged from 6.0E-8 to 1.0E-6 per reactor year. This is approximately 100,000 times more probable than the analytically estimated probability of a double ended guillotine pipe break. Even though estimates of the probability of pipe ruptures and leakage may be inexact, leakage is much more probable and will precede a pipe rupture.

4.5.2 Mitigation of Intergranular Stress Corrosion Cracking

NUREG/CR-4792 reported a higher pipe break failure probability of 1.0E-3 per reactor year due to IGSCC. This was based upon conservative assumptions regarding stress conditions. This probability is considered inappropriate for use in assessing the safety significance of potential ECCS suction strainer blockage for the following reasons:

- No credit is taken for the probability that cracks would be detected through in-service inspection prior to failure.
- It does not give any credit for the potential benefits of alternate mitigative actions such as pipe replacement, weld overlay, mechanical or induction heating stress improvement, hydrogen water chemistry, zinc addition, etc. Mitigative actions have been taken by all operating BWRs in response to NRC Generic Letter 88-01 [14] and NUREG-0313, Revision 2 [15].
- Reference 7 demonstrates that 1E-3 per reactor year is not an appropriate basis.
- Detectable leakage would still be expected in stainless steel pipes prior to rupture.

4.6 Leak-Before-Break

4.6.1 Piping Failure Modes

Industry experience indicates that high energy pipes experience leaks long before a pipe break condition develops. This is referred to as "Leak-Before-Break". The concept of Leak-Before-Break (LBB) is that large ruptures of austenitic stainless steel piping (BWR recirculation systems) are extremely unlikely without a preceding period of leakage that would be detectable. This would allow for plant shutdown and repair prior to gross failure of reactor coolant piping.

Although the design basis for nuclear power plants includes the evaluation of a LOCA resulting from a postulated pipe break, considerable effort is applied towards designing piping and safe-end systems to assure that such a break will not occur. Piping systems are analyzed using appropriate codes and standards to limit applied stresses. Materials are selected to provide adequate ductility and toughness. Piping design also provides implicit safety margin concerning fatigue initiation.

Even if a pipe or safe-end should experience cracking that was not detected by in-service inspection, the crack would grow to a through-wall leak and the leak would be detected well before it reaches critical crack size. The critical crack length is that length at which pipe rupture is predicted. It is a key paramet in the consideration of LBB. This critical crack basis already exists in most plant ; Analysis Reports (SAR) as part of the plant design basis.

The LBB concept is based on the fact that reactor piping and safe-ends are fabricated from tough ductile materials that can tolerate large through-wall cracks without complete fracture under service loading. By monitoring the leak rate from through-wall cracks, leakage from piping can be detected well before the margin to rupture is challenged. With leak detection systems and conservative limits on allowable leakage, cracks can be isolated and/or an orderly plant shutdown initiated prior to the occurrence of a large pipe rupture.

4.6.2 Critical Crack Length

It is important to understand the leakage rate that will occur prior to achieving the critical crack length. The critical crack lengths and leak rate for typical BWR piping geometries have been documented in plant SARs and Reference 16. These documents show that the calculated leak rate at the critical crack length is a strong function of pipe diameter with the LBB margin increasing with increasing pipe size. Thus, larger pipes where failure would be more significant for insulation debris generation, have significant early warning advantages due to LBB. While the LBB margin is somewhat lower for smaller pipes, there is still a large BWR experience database supporting the integrity of such piping.

For larger diameter piping, a detectable leak rate is expected well before the crack grows to the critical crack length. Even for a line as small as a 4 inch diameter water line, the predicted water leak rate is about 25 gpm at the critical crack length. (Table 2 of Reference 16 gives typical leakage rates versus pipe size.) The leak detection system sensitivity is such that alarms occur around 5 gpm which is well before the critical crack length.

The BWR piping systems are expected to develop detectable leaks long before reaching the point of an incipient rupture. Therefore, the assumption of a leak-before-break type failure scenario is appropriate for evaluating BWR piping systems relative to ECCS suction strainer performance.

4.6.3 Leak Detection Monitoring

The leak detection system in the drywell, where debris generation from a pipe break could potentially impact the ECCS suction strainers, is capable of detecting small unidentified leaks. This is accomplished by monitoring of the drywell floor and equipment drain sump water level/pump operation, and airborne gaseous radioactivity increases. These variables are continuously indicated, recorded, and alarmed in the main control room.

The Technical Specifications require plant shutdown upon identification of any primary system pressure boundary leakage [14]. To monitor this, instrumentation is set to alarm when unidentified leakage reaches 5 gpm. Unidentified leakage is limited to 5 gpm by the Technical Specifications. This assures that through-wall cracks in piping within the drywell will be identified early. Additionally, plant Technical Specifications limit the increase in unidentified leakage to 2 gpm over a specified time (typically 8-12 hours). Plant operators closely monitor any increase in unidentified drywell leakage. Finally, total identified leakage in the drywell is limited by the Technical Specifications to approximately 25 gpm.

If the floor drain fill/pump-out system detects leakage exceeding this total leakage limit, it alarms in the main control room. If either unidentified leakage or total identified leakage exceeds the Technical Specification limits, the plant must enter a Limiting Condition for Operation (LCO) which requires correction. If not corrected within the specified time, the plant must be depressurized and placed in a shutdown condition. This would be well before reaching leakage rates that would effect strainer performance.

As stated, the Technical Specifications do not allow continued operation with any primary system pressure boundary leakage and the leak detection monitoring system assures that unidentified leakage is brought to the attention of the operating staff and corrected. This assures early detection of any leakage before design basis accident conditions can develop.

4.6.4 Leak-Before-Break Application to the ECCS Suction Strainers

The Design Basis Accident or the DEGB design requirement was originally conceived to provide a deterministic basis for margin in plant design. It was felt that this would be a conservative basis for sizing plant equipment. NUREG-1061, Volume 3 [17] stated this as follows:

> *"The "design basis accident", "maximum credible accident" or "maximum hypothetical accident" have been used as terms describing what was generally the double-ended guillotine break. The concept was originated by the U.S. Atomic Energy Commission for the multiple purpose of sizing containments and establishing "accident" doses and later, the sizing of emergency core cooling systems. The original concept was quite straightforward: namely, an instantaneous DEGB of a major pipe in the primary system of a light water reactor (LWR) would maximize the fluid release and establish an upper bound for the design pressure established for the containment. This optimized the containment volume vis-à-vis a reasonable design accident pressure."*

In addition, NUREG-1061 states that the probability of a DEGB is extremely low. While it did not recommend any change in the design basis for system sizing, it did establish the principle of secondary effects of the DBA that did not need to be designed to the full accident conditions. It stated:

> *"Operating experience suggests that leak-before-break is the most likely mode of failure for the vast majority of cracks occurring in service. This is a result of the asymmetry of weld residual stresses and applied loads and the variability in material properties. Evaluations using conservative crack growth rate predictions and net section collapse analyses, applicable to cracks in very high toughness material indicate that for the vast majority of possible crack geometries there exists significant time to allow for detection of leakage and implementation of corrective actions. Evaluation using the fracture resistance properties of weld material show substantial margins against failure, under normal and accident loading conditions for throughwall cracks which should be reliably detected by leakage."*

As a result of the evaluation performed by the NRC Piping Review Committee, use of LBB considerations was allowed to resolve USI A-2: Asymmetric Blowdown Loads on PWR Primary Systems. This avoided costly backfit of pipe whip restraints and consideration of annulus pressurization and jet impingement loads in the plant design. To allow limited further application of the secondary effects of jet impingement the NRC revised General Design Criteria (GDC) 4 as follows:

> *"However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."*

The application of LBB to the issue of ECCS suction strainer blockage would limit the design basis for the strainers to minor insulation debris from pipe leakage. Debris generated from such leakage should be well within the capability of the strainers. Jet impingement caused by pipe rupture is a secondary effect of the DBA and the limited application of LBB to the ECCS suction strainer issue would appear to be a potential consideration in light of the other mitigative actions already taken to address IGSCC. In the question and answer responses to NUREG-0869 [18], the NRC indicated that they would consider LBB for individual plant applications to resolve USI A-43.

4.7 Pipe Break Core Damage Frequency

4.7.1 Core Damage Frequency Impact

For insight, it is useful to look at the large break accident (DEGB) from the standpoint of its impact on core damage frequency (CDF).

Because the insulation debris does not distribute uniformly to the pool and the strainers are located around the pool, the simultaneous blockage of the ECCS suction strainers is considered to be an extremely unlikely scenario. From Section 4.5.1 the pipe break probability determined analytically for a large recirculation pipe break was estimated to be approximately 1.0E-12 events per reactor year. On the basis of operational experience estimates, it was estimated to be 7.51E-6 per reactor year. Because the "upper bound" operational experience estimate was based upon a history of no prior large breaks, a best estimate conclusion might be that the actual failure probability would be several orders of magnitude lower than the upper bound. Therefore, it would not be unreasonable to assume the probability of a large break to be <1.0E-6 per reactor year. Placing the value at the high end allows some margin for factors such as erosion and water hammer.

Typical Individual Plant Evaluations (IPEs) show that the design basis accident is not a major contributor to the overall plant core damage frequeny. With a large pipe break frequency of <1.0E-6 per reactor year, even if the ECCS suction strainers were assumed to fail during a DBA, the CDF could not be above the initiating event frequency. The NRC safety goal [19] policy contains the following statement:

> *"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."*

This is also consistent with American Nuclear Society Standard 52.1 [20]. Using the above best estimate result of <1.0E-6 per reactor year for a large pipe break frequency would allow further actions in response to this concern to be based upon severe accident considerations.

4.7.2 CDF Estimate for a Large Break LOCA

An estimate was made for the CDF due to strainer blockage from fibrous debris generated for a large break LOCA. It assumed a very high probability of pipe failure and ECCS suction strainer blockage, but utilized a realistic evaluation taking credit for other mitigation systems. The assumptions for this hypothetical case follow:

- The LOCA frequency is conservatively assumed to be 1.0E-4 per reactor year for large breaks and 1.0E-3 for small and medium size breaks.

- Any size LOCA, not just the DEGB, would dislodge insulation from the pipes.

- The insulation from any size break is transported to the suppression pool and is instantaneously deposited on all of the ECCS pump suction strainers. The low pressure ECCS pumps are totally disabled, i.e., the probability of blockage is 1.0 for the assumed event.

- The BWR 5/6 High Pressure Core Spray (HPCS) make-up system initially takes suction from the CST and it can provide adequate core cooling for any size LOCA.

- Low Pressure Core Spray (LPCS) pumps on BWR 2/3/4 plants that can be manually aligned to the CST can mitigate any size breaks after depressurization.

- High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems can mitigate small and medium LOCAs, but would be unavailable for the large break.

- Other systems available to mitigate the large breaks are the condensate pumps, RHR-Service Water (RHR-SW) cross tie, and the firewater system pumps. (See Section 4.4.)

The HPCS and HPCI systems are initiated automatically. Typical unavailability for these automatically initiated systems is 1.0E-2 per demand. The LPCS realignment to the CST and the other systems available to mitigate large breaks require operator action that is assumed to dominate their unavailability. Because of potential dependency of the operator actions, LPCS is assigned a rather high unavailability of 1.0E-1 per demand. Similarly, the alternate sources of water (condensate, RHR-SW cross tie, and fire water systems) are also assigned a high unavailability of 1.0E-1 per demand.

The availabilities of these alternate sources of water are not affected by ECCS suction strainer blockage and there is very little potential for common cause failure that can disable all of these alternate systems. However, as stated above, potential dependency among the systems requiring operator action has been accounted for by assigning a relatively high system unavailability.

The CDF for a large break LOCA for a BWR 5/6 plant is estimated as follows:

CDF = (LOCA frequency)(HPCS failure)(RHR-SW failure)(Firewater Failure)

CDF = (1.0E-4)(1.0E-2)(1.0E-1)(1.0E-1)

CDF = 1.0E-8 per reactor year

For BWR 2/3/4 plants the HPCI cannot mitigate the large break, but the LPCS can be manually aligned to the CST. The estimated CDF for a large break LOCA for these plants is 1.0E-7 because you replace an automatic initiation with a manual initiation by the operator. This CDF number might change to 1.0E-6 when calculated for plant specific conditions (e.g., no RHR-SW cross tie).

The above estimate did not take any credit for the condensate injection because it is not able to mitigate some specific breaks. For those breaks that condensate injection can mitigate, the estimated CDF would be an order of magnitude lower. The estimate also did not account for seismic effects. Because of the high seismic capacity of the reactor piping, the probability of a seismic induced LOCA is much lower than that used in the above estimate. For seismic events, even with the failure of non-safety CSTs, the CDF is expected to be lower than those given above.

For small break LOCAs, even though the break frequency is higher (1.0E-3 per reactor year), the automatic initiation of the Reactor Core Isolation Cooling (RCIC) that is aligned to the CST would help mitigate this event. Plants without RCIC may have other mitigating systems such as CRD pumps powered by the diesel. The estimated CDF for the small break LOCA is lower than for the large LOCA.

This estimated CDF value is low enough (<1.0E-6) that further actions to mitigate the postulated ECCS strainer blockage event should be allowed to be based on severe accident considerations. Additionally, if the experience based estimates for the LOCA frequency were used (7.51E-6), the CDF result would be two orders of magnitude lower.

5.0 Conclusions

This safety assessment concludes that continued operation of U.S. BWRs is acceptable while each plant evaluates its compliance to 10CFR 50.46 and implements any corrective actions that are required. This conclusion is based upon the following:

- In the improbable event of a design basis accident, the response of U.S. BWRs would be expected to be different from the Barsebäck event. The Barsebäck event itself is precluded in U.S. BWRs by the piping of the pressure relief function directly to the suppression pool. The hypothetical DEGB could cause insulation debris to be generated. However, the Nukon™ and reflective metallic insulation used in most U.S. BWRs would be expected to be much less susceptible to damage and transportation than the mineral wool employed at the Barsebäck plant. In most cases, containment design differences from Barsebäck would provide more of a barrier to transportation of insulation from the drywell to the suppression pool. Insulation properties and pool velocities should, in most cases, minimize the transportation within the pool and the subsequent blockage of the strainers. While each plant is different in the containment layout and insulation design, U.S. BWRs are expected to be less susceptible than Barsebäck to strainer blockage.

- Actions taken in response to Bulletin 93-02 will help to decrease the potential for the introduction of foreign materials into the suppression pool. By keeping the containment clean and foreign material out of the pool, it will lessen the potential for the combined effects of blockage from sources other than insulation.

- Even assuming the occurrence of the DEGB, the operator will recognize the effects of strainer plugging and has available actions to maintain adequate core cooling in accordance with plant EOPs. This includes utilization of alternate water injection systems and sources. The EOPs would lead him through the required action even without specific recognition of the blockage of the strainers. Actions taken in response to NRC Bulletin 93-02, Supplement 1 will further raise the operator's awareness of this potential event and possible actions to help mitigate strainer blockage events.

- The low probability of a large pipe rupturing, as demonstrated by operational experience and analytical techniques, would indicate that the most probable mode of pipe failure is leakage. This leakage would be detected by currently installed plant instrumentation and the operator would be required by the Technical Specifications to take action to shutdown the plant if leakage exceeded specified limits. The leak, which would not introduce any concern regarding strainer blockage, would then be repaired. The NRC accepted the principle of leak-before-break in resolving USI A-2 (Asymmetric Blowdown Loads on PWR Primary Systems). If that were extended to the ECCS suction strainer issue, there would be no jet impingement forces and no significant debris of concern.

In order to assess compliance to 10CFR 50.46 it is appropriate to develop an updated Regulatory Guide 1.82 evaluation approach. The model for insulation debris generation, transport, and head loss would vary for each containment type and insulation type. The BWROG is developing evaluation methodologies to assist the owners' in their specific plant evaluations and is also exploring long term fix options for those who might require them.

6.0 References

1. Regulatory Guide 1.82, Rev. 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident", November 1985

2. Licensee Event Report 93-011-00 for the Perry Nuclear Power Plant, Unit 1, May 1993

3. NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers", May 11, 1993 & Supplement 1, February 18, 1994

4. Licensee Event Report 93-007-00 for the Susquehanna Steam Electric Station, July 1993

5. Residual Heat Removal System, GE Design Specification, 22A3139, Revision 5, February 1, 1977

6. The Barsebäck 2 Incident and the Role of the Thermal Insulation, by Gordon H. Hart, P.E., October 1993

7. Nukon™ Nuclear Containment Insulation System ECCS Performance, Section 4, Performance Contracting Inc., March 27, 1985

8. Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps", November 2, 1970

9. NRC Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment", April 1993

10. Operator Response to a Loss of ECCS, BWR Owners' Group Report, November 1993, Transmittal BWROG-93149

11. Emergency Procedure Guidelines, Rev. 4, NEDO-31331, March 1987

12. NUREG/CR-4792, Volume 1, "Probability of Failure in BWR Reactor Coolant Piping", Vol. 1: Summary Report, March, 1989

13. "Pipe Break Probabilities in Boiling Water Reactors", BWR Owners' Group Report, November 1993, Transmittal BWROG-93149

14. NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping", January 1988

6.0 References (cont.)

15. NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", Final Report, January 1988

16. Safety Assessment, "Isolation of MOVs for HPCI and RCIC Steam Supply Line and RWCU Water Supply Line", Rev. 1, August 1990

17. NUREG-1061, Vol. 3, "Evaluation of Potential for Pipe Breaks", November 1984

18. NUREG-0869, "USI A-43 Regulatory Analysis", October 1985

19. NRC 10 CFR Part 50, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Correction and Republication", August 1986

20. Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, American Nuclear Society, ANSI/ANS-52.1-1983

BWR THERMAL INSULATION TYPES AND ECCS CHARACTERISTICS

Plant	Type of Insulation in Drywell	ECCS Pumps (1)	Pumps per Strainer	TotalStrainer SurfaceArea (sq. ft.)	Total Strainer Suction Flow (gpm)	FSAR NPSHA (ft.) (2)	FSAR NPSHR (ft.)
Big Point Rock (No supp. pool)	Nukon Asbestos Cal-Sil Blocks	RHR-2	5 Strainers Total	3.7	400	24.5	23
Browns Ferry 1,2,3	Reflective Metal	RHR-4 CS-4	4 Strainers Total	40.0	10,000 3,125	- -	- -
Brunswick 1,2	Nukon Reflective Metal	RHR-4 CS-2	2 1	32.2 15.7	15400 4625	- -	15 12
Clinton	Reflective Metal	RHR-3 LPCS-1 HPCS-1	1	157.0	31,030	14.2 12.8 11.7	5 5 5
Cooper	Reflective Metal	-	-	-	-	-	-
Dresden 2,3	Reflective Metal limited Nukon	LPCI-4 CS-2	4 Strainers Total	-	5000 4500	42 42	30 28
Duane Arnold	Nukon	RHR-4 CS-2	2 1	14.6 4.21	9600 3020	24 32	- -
Fermi 2	Reflective Metal some Fiberglass	RHR-4 CS-4	1 2	-	10000 6350	20.3 21.2	16.8 17.2
Fitzpatrick	80% Nukon 20% Mineral Wool	RHR-4 CS-2	2 1	26.7 11.9	20800 4725		- -
Grand Gulf	Reflective Metal Fiberglass K-Wool, Min-K Ca Sil w/Jacket	RHR-3 LPCS-1	Ea. Pump has 2-100% Strainers	31.25 Open Area 125 Total Area	22,350 7115	5.5 6.4	2.0 1.75
Hatch 1	Reflective Metal Fiberglass Calcium Silicate	RHR-4 CS-2	1 1	34.4 9.0	30,800 9,400	26.0/33.6 23.4/31.0	15.2 10.2
Hatch 2	Nukon Reflective Metal Fiberglass Calcium Silicate	RHR-4 CS-2	1 1	78.0 28.2	30,800 9,400	10.2/21.6 8.6/17.9 (16)	6.3 7.3
Hope Creek	Nukon	RHR-4 CS-4	1 1	39 open area 14 open area	10500 4015	10.6 11.2	9 10
LaSalle 1,2	Reflective Metal	RHR-3 LPCS-1	1 1	16.68 16.68	8400 8100	25 25	15.5 2
Limerick 1,2	Nukon some Fiberglass	RHR-4 CS-4	1 1	27.7 13.3	10000 3715	17.5 12	5 6.5

ATTACHMENT 1

BWR THERMAL INSULATION TYPES AND ECCS CHARACTERISTICS

Plant	Type of Insulation in Drywell	ECCS Pumps (1)	Pumps per Strainer	Total Strainer SurfaceArea (sq. ft.)	Total Strainer Suction Flow (gpm)	FSAR NPSHA (ft.) (2)	FSAR NPSHR (ft.)
Millstone 1	Primarily Reflective Metal Some Fiberglass	RHR-4 CS-2	6 Pumps 3 Strainers	71.1	27200	-	-
Monticello	Nukon on pipes Mirror Insulation on RPV	RHR-4 CS-2	6 Pumps per 4 Strainers	40.1 (15)	22040	31	28
Nine Mile 1	Reflective Metal some Fiberglass	RHR-4 CS-4	1 1	17.5 Tot 14.5 Flow	- -	35.4 37.0	35.2 35.0
Nine Mile 2	Reflective Metal limited Fibrous	RHR-3 LPCS-1	1 1	14 ea 8.9 ea	8200 7800	15.1 11.2	14.0 11.2
Oyster Creek	Primarily Nukon Reflective Metal on RPV	LPCS-4 RHR-4 (4)	3 Common Strainers (4)	(6) 55.5 Tot Strainers 24.4 Total Flow	(5) 11400 gpm- 3 Strainers	23.11	16.5
Peach Bottom	Nukon	RHR-4 CS-4	1 1	44.4 30.8	10000 3125	- -	- -
Perry	Primarily Nukon Reflective Metal on RPV	RHR-3 LPCS-1 HPCS-1	1 1 1	211 Total	39,000 Totals	6.0 6.9 5.3 (7)	4.0 4.0 4.0 (13)
Pilgrim	Nukon	RHR-4 CS-2	1 1	13.3 13.3	5250 5250	30 -	- -
Quad Cities	Reflective Metal some Nukon	RHR-4 CS-2	4 Strainers Total	-	5000 4500	42 42	30 28
River Bend	Fiberglass Nukon Min-K Temp-Mat	RHR-3 LPCS-1	1 1	70.8	20,160	4.0 4.76	0.3 0.3 (14)
Susquehanna	90% Reflective 10% Nukon(8)	RHR-4 CS-4	1 2	211.8 Total (10)	61,500 Total	20.8 11.6 (9)	7 11
Vermont Yankee	Nukon some Fiberglass Reflective Metal	RHR-4 CS-2	2 1 (12)	71.7	17,400	33 34	26 24

ATTACHMENT 1

BWR THERMAL INSULATION TYPES AND ECCS CHARACTERISTICS

NOTES:

(1) RHR = LPSI and/or containment spray pumps; CS = Core spray pumps; LPCS = Low pressure core spray; Also HPCS for those applicable plants

(2) $NPSH_A$ = available net positive suction head; $NPSH_R$ = required NPSH

(3) Information which is currently unavailable is indicated by a dash.

(4) Only 2 out of 4 LPCS can run simultaneously.

(5) Design Basis Flow Rate

(6) $\frac{\text{Total Strainer Area}}{\text{Total Flow Area}}$

(7) $NPSH_A$ = Conservatively includes a 4.0 psid strainer fouling limit. The actual strainer fouling ΔP is less than 1.0 psid @ 80% fouled

(8) Unit 1 has ≅ 2–3% Nukon (Nukon is being replaced in approximately 75% of the locations)
Unit 2 has ≅ 10% Nukon (2–3% after upcoming spring outage modification)

(9) It should be noted that SSES takes a penalty for high temperature (200°F); but does not take credit for high wetwell pressure (≅16.4 psi) during the long-term accident response. Realistic values for NPSHA (assuming no head loss across strainer).
RHR 32.9 ft.
CS 32.1 ft.

(10) The strainer surface area is based upon total surface area, rather than flow area (hole area). Per strainer vendor design, flow area is approximately 40% of total surface area.

(11) Best estimate NPSH for WNP-2 is 72 ft. for all pumps.

(12) For VY each suction inlet has a tee with two strainers. There is one suction inlet for each CS pump and one suction inlet for each pair of RHR pumps.

(13) For Perry the realistic NPSH is RHR-29.3', LPCS-30' and HPCS-28.5' and Perry utilizes Post Accident Peak Suppression Pool Water temp rather than 212°F and actual 50% fouled strainer ΔP.

(14) For River Bend these NSPH values are adjusted to a reference elevation of 3' above pump mounting flange as specified in the performance test from pump vendor.

(15) For Monticello this represents combined strainer area for 4 strainers connected to a common ECCS suction ring header.

(16) For Plant Hatch:

- Strainer Area is total surface of the strainers for all pumps. For example, all four Unit 2 RHR pumps have a combined strainers surface of 78.0**2. Flow area is about 40% of this number (i.e., the holes make up about 40% of the strainer area).
- Strainer suction flow is the total rated (not run out) flow for all pumps. For example, all four Unit 2 RHR pumps have a combined rated flow of 30,800 gpm. Plant Hatch procedures instruct operators to throttle pumps to rated flow, so this appeared more appropriate than run out flow.
- NPSH-Required is provided at rated rather than run out conditions, as discussed above.
- NPSH-Required is given for both "licensing" and "realistic" conditions. Unit 1 is pre-Reg Guide 1.1 plant, and does take credit for post-LOCA containment pressure for NPSH in the FSAR. Unit 2 is licensed to RG 1.1, and licensing NPSH-Required values do not consider containment pressure. GE reports on suppression pool pressure and temperature response were used to calculate "licensing" and "realistic" NPSH-Required values.

OPERATOR GUIDANCE FOR POTENTIAL BLOCKAGE OF ECCS PUMP SUCTION STRAINERS

The purpose of this document is to alert plant operators to the potential of a common mode failure of ECCS and containment cooling systems which take suction from the suppression pool. This may occur during a Loss-of -Coolant Accident (LOCA) in the drywell. The force of the steam/fluid mixture escaping from the break disrupts insulation on nearby piping and equipment, generating debris. A fraction of the debris generated is transported to the suppression pool. ECCS operation results in deposition of debris on the suction strainers. The pressure drop increases across the strainers decreasing available net positive suction head (NPSH) at the ECCS pump suction or cutting off the flow altogether, resulting in loss of ECCS. The ability to cool the core may be lost and core damage may occur if other sources of injection are inoperable or incapable of injection. The ability to protect the containment may also be lost for RHR and other systems which take suction from the suppression pool.

When Suction Strainer Blockage Might Occur

INPO Significant Event Notification 90 (SEN-90) describes the event which occurred at Barsebeck, where an unpiped safety valve opened and discharged to the containment. Containment sprays actuated and swept debris to the suppression pool. The strainers required backwashing after about 1 1/2 hours into the event. SEN-90 states that calculations indicate these strainers could become blocked in less than 30 minutes. Therefore, the need for actions to line up alternate injection systems could be required fairly early in an event.

Indications of ECCS Suction Strainer Blockage

ECCS suction strainer blockage may be detected through one or more indications of degraded system performance. As the blockage progressively increases, these indications may include:

System flow rate less than expected for the backpressure to which the system is discharging (i.e., RPV, drywell, or suppression pool pressure).

Decreased suction pressure. For most plants the ECCS suction pressure indications are local.

Decreased pump motor current indications. For plants equipped with ECCS pump motor ammeters (local indications only for some plants), current indications decrease with the reduction in system flow.

Frequent unanticipated adjustment of system discharge valve (for those plants which have throttle capability in the control room). For example, given a steady state conditions is reached, the discharge valve must be periodically adjusted to increase flow.

Inability to control and maintain parameters such as RPV water level, containment pressure, drywell temperature, suppression pool temperature and suppression pool level within the bounds of specified EOP action levels and limits.

Abnormally low discharge pressure indication for a given flow: As the suction strainer blockage restricts flow to the pump, suction pressure decreases which produces a lower than normal pump discharge pressure for a given flow rate;

Erratic and dramatic fluctuations in discharge pressure, flow, and pump motor current indications; indications of cycling of the minimum flow valve; discharge pressure low alarms indicative of a loss of ECCS keep-fill.

Minimum flow valve open indication: Increased suction strainer blockage may reduce discharge flow causing the minimum flow valve to open.

Most of the indications of degraded system performance are system-specific requiring the comparison of current values and trends to data obtained when the system is known to be functioning properly. Early recognition of suction strainer blockage would be dependent on the observations made by the operators as they place the systems in service and adjust system flows to meet the requirements dictated by the EOPs.

EPG/EOP Actions

Plant EOPs developed from Rev. 4 of the BWROG EPGs specify diverse and redundant systems for controlling RPV water level. Since the symptom-oriented EOPs must address a full spectrum of initial plant conditions and postulated transients, the EOPs do not unconditionally prioritize use of one injection source over another. Sources of RPV injection include those systems used to control RPV water level during normal plant operations at power (e.g., feed and condensate, CRD, etc.) and those categorized as emergency makeup (e.g., high and low pressure ECCS, Alternate Injection Subsystems, etc.). Alternate injection subsystems vary from plant to plant but typically include the RHR service water crosstie, fire system, interconnections with other units, the ECCS keep-full system, and others.

The EPGs permit the use of Alternate Injection Systems upon entry to the water level section of the RPV Control Guideline and in several EPG contingencies:

Contingency # 1, Alternate Level Control, provides guidance for the use of alternate injection subsystems to help reverse a decreasing water level trend. This guidance is entered from the RPV Control Guideline when the operator determines that RPV water level cannot be maintained above the top of the active fuel (TAF).

Contingency #4, RPV Flooding, provides guidance for the use of Alternate Injection Subsystems to help establish an RPV pressure for a given number of open SRVs; thereby, assuring adequate core cooling when RPV water level cannot be determined.

Contingency #5, Level/Power Control, provides guidance for the use of Alternate Injection Subsystems when reactor shutdown cannot be assured.

Contingency # 6, Primary Containment Flooding, provides for injection when the RPV level cannot be restored and maintained above the TAF or adequate core cooling cannot be assured and specifies injection from sources external to the primary containment (i.e., other than the suppression pool) with all available injection.

The Primary Containment Control Guideline specifies operation of RHR and other systems modes which spray and/or cool the containment and reject water from the suppression pool.

Mitigative Actions

Blockage of the ECCS suction strainers may occur due to debris created from a LOCA in the drywell and transported to the suppression pool. In addition to any debris resident in the drywell and suppression pool prior to the LOCA, the amount of debris which reaches the pool is a function of the path it must take through the drywell and vent header system. The deposition of the debris on the ECCS suction strainers is a function of its material composition, the sink rate, the strainer size, and the suction flow which entrains the debris. Consequently the potential mitigative actions are both plant-specific and event-specific. Within the latitude provided by a plant's EOPs to restore and maintain parameters within specified limits, potential mitigative actions may include:

Minimizing ECCS division flow: ECCS divisions not needed to restore and maintain EOP parameters within specified limits should be removed from service; if EOP instructions do not require full division flow, for those plants with capability to throttle ECCS, the inservice division should be throttled to meet the flow demanded by the EOPs (note: prolonged operation of ECCS pumps on minimum flow should be avoided). These actions may reduce the entrainment of debris and deposition on the suction strainers, and thus may prolong the operability of the inservice ECCS division.

Alternating ECCS divisions (for plants which have one strainer per division or one strainer per pump): After ECCS division flow is minimized and if the inservice ECCS division performance degrades such that EOP parameters cannot be restored and maintained to within specified limits, replace the degraded in-service division with a standby ECCS division. These actions may reduce the entrainment of debris and deposition on the suction strainers, thus prolonging the time ECCS maintain EOP parameters within specified limits.

Shifting ECCS suction: Where event-specific conditions permit, the source of suction for one or more ECCS divisions should be transferred from the suppression pool to the CST or other suction source outside primary containment. This action will reduce and may prevent the entrainment of debris and deposition on the suction strainers as long as CST suction is available, thus prolonging the time ECCS maintain EOP parameters within specified limits.

Timely operation of Alternate Injection Subsystems: When available systems and subsystems are inoperable or incapable of injecting into the RPV, alternate injection subsystems should be lined up and placed in service as directed by the plant EOPs.

Venting the primary containment if containment cooling is lost and cannot not be restored, as directed by the plant EOPs.

Recommendations

Ensure plant operators are familiar with the expected performance of ECCS systems when operated in the RPV injection mode and, where applicable, suppression pool cooling and containment spray modes. This includes recognition of nominal system parameters such as discharge flows and pressures, motor current indications, suction pressure, pump noise and vibration, minimum flow valve opening and closing flow rates, etc.

Ensure operators are familiar with expected performance of Alternate Injection Subsystems and requirements for placing them in service. This includes recognition of maximum RPV injection pressures, expected RPV injection flow rates, sources of injection, injection flowpaths, resource and time limitations impacting subsystem lineup, etc.

Ensure supporting operating procedures provide sufficient flexibility so that possible mitigation actions can be effectively performed (e.g., lineup of ECCS divisions to the CST during emergencies should be permitted in the plant operating procedures).

Ensure plant operators are cognizant of the latitude provided in EOP decisions and actions related to the operation of RPV injection systems, Subsystems, and Alternate Injection Subsystems. This includes recognition of:

the option to augment RPV injection with Alternate Injection Subsystems while controlling level in the water level control section of the EOP, and

the need to enter the EOP developed from EPG Contingency #1, Alternate Level Control, when the determination is made that RPV water level cannot be maintained above the top of the active fuel (i.e., the transition to Contingency #1 need not be delayed until RPV water level reaches TAF).

Future Actions

The BWROG Emergency Procedure Committee (EPC) will continue to review this issue. This paper should not be construed to authorize a change to the BWROG EPGs. The EPC will provide notification to you if any EPG changes are identified and approved to address this issue.

ATTACHMENT 2

References:

NRC Bulletin No. 93-02, Supplement 1: Debris Plugging of Emergency Core Cooling Suction Strainers, February 18, 1994

NRC Bulletin No. 93-02: Debris Plugging of Emergency Core Cooling Suction Strainers, May 11, 1993

NRC Information Notice 93-34: Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment, April 26, 1993

INPO Significant Event Notification 90, Blockage of Containment Spray Suction Strainers, dated September 29, 1992

NRC Information Notice 92-71: Partial Plugging of Suppression Pool Strainers at a Foreign BWR

NRC Information Notice 88-28: Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage

INPO Significant Event Notification 31, Delamination of Drywell Insulation Material, dated March 31, 1988

eccs.sct
3/15/94