April 28, 2006

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U.S. Nuclear Regulatory Commission
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Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Supplemental Response to Generic Letters (GL) 98-04 and GL 2004-02
Licensee Event Report 266/301/2005-006-00

References:
1. Letter from WEPCo to NRC dated November 11, 1998 (NPL 98-0950)
2. Letter from NMC to NRC dated September 1, 2005 (NRC 2005-0109)
3. Licensee Event Report 266/301/2005-006-00, dated January 9, 2006
4. NRC Request for Additional Information (RAI) dated January 10, 2006, Regarding Event Notification 42129 (TAC Nos. MC9035 and MC9036)
5. Letter from NMC to NRC dated February 16, 2006 (NRC 2006-0009)

The purpose of this letter is to provide supplemental information as committed to in Licensee Event Report 266/301/2005-006-00 (Reference 3). Reference 3 committed to supplement the Licensee's response to Generic Letters 98-04 and 2004-02 that was associated with emergency core cooling system (ECCS) degradation following a design basis accident (DBA). The original commitment date for this submittal was April 15, 2006. A telephone conversation between the Nuclear Regulatory Commission (NRC) Project Manager for Point Beach Nuclear Plant (PBNP) and the plant staff extended this submittal date to April 28, 2006.

On November 8, 2005, NMC submitted Event Notification 42129 in accordance with the requirements of 10 CFR 50.72(b)(3)(ii)(B) and 10 CFR 50.46(b)(5). The notification reported the identification of errors in the calculations that were used as a basis for responding to NRC Generic Letter GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment." The errors were in three distinct areas, and each error was non-conservative. Two operability evaluations and a supporting calculation were subsequently performed to demonstrate adequate net positive suction head (NPSH) would be available to the (ECCS) pumps to ensure long-term cooling, and that air entrainment would not occur. LER 266/301/2005-006 (Reference 3) was submitted on January 9, 2006. This LER provides the results of NMC's investigations and conclusions regarding this issue. The operability evaluations and supporting calculation were submitted to the Commission via Reference 5 and were discussed in that letter.

On January 10, 2006 (Reference 4), a request for additional information (RAI) was issued by the NRC staff regarding the event notification made on November 8, 2005. NMC responded to this RAI via letter dated February 16, 2006 (Reference 5). Enclosure 2 of the NMC response of February 16, 2006, is included with this letter, also identified as Enclosure 2, to facilitate NRC staff review in accordance with a conference call held on April 11, 2006, between NMC and NRC representatives. During that conference call, a newly discovered condition associated with the potential for remote operation of the containment sump isolation valves during a postulated low or degraded voltage condition was discussed. NMC has performed an operability recommendation associated with that discovery. Evaluations of the condition are continuing. NMC is also reviewing information provided via Reference 5 in light of this discovery, and will supplement that response, if appropriate.

Enclosure 1 provides supplemental information to GL 98-04 and GL 2004-02 in fulfillment of the commitment made in Reference 3.

Summary of Commitments

There are no new commitments or revisions to existing commitments contained in this letter.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on April 28, 2006.

Dennis L Koehl

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures (2)

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC

ENCLOSURE 1
SUPPLEMENTAL INFORMATION
GENERIC LETTERS (GL) 98-04 AND GL 2004-02
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In Licensee Event Report (LER) 266/301/2005-006-00, "Calculation Errors in Model for ECCS Long Term Cooling," NMC committed to supplement its responses to NRC Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"; and to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents and Pressurized-Water Reactors." The supplemental information follows.

GL 98-04 Supplemental Information

Coatings

The Licensee's original response to GL 98-04 was transmitted in Reference 1. During NMC's review of this generic communications response in conjunction with the net positive suction head (NPSH) calculation concerns documented in Reference 3, it was discovered that the precautionary measures recommended by the consultant in 1989 for Unit 1 and 1990 for Unit 2 to minimize the possibility of blockage in the near-sump zone had not been implemented. The recommendations that had been made in the 1989-1990 studies performed by the consultant which were discovered to not be implemented were:

Unit 1: (1) Removal of concrete coatings from the reactor in-core instrumentation access shaft wall adjacent to the sump screens; (2) removal or installation of a canopy for top coat from steel surfaces in the near sump screen area up to the ceiling and up to eight feet all round the sump screens

Unit 2: (1) Removal of concrete coating from the "B" steam generator compartment exterior walls adjacent to the emergency sump screens up to the ceiling at El. 8'; and (2) removal of steel coatings within eight feet of the sump screens up to the ceiling at El. 8' with application of one coat of Dimetcote 6 inorganic zinc primer, or installation of a canopy on top of the sump screens.

The recommendations made in these reports conservatively assumed all coatings in containment, regardless of service level, would fail under post-LOCA conditions. In that context, any coatings in the immediate vicinity of the ECCS sump screens could jeopardize their operation by being transported to the screen surface prior to settling to the containment floor.

However, the coatings referred to (primarily on the containment liner and on nearby vertical concrete surfaces) were considered "qualified" coatings. As such, they are not presumed to fail under post-LOCA conditions, and would not be a challenge to the operation of the sump screens. Therefore, removal of the coatings or installation of a canopy would be necessary only if the station sought to downgrade the service level of the coatings. The option of downgrading the service level of the coatings was not pursued by the station.

Enclosure 2, Attachments 1 and 2 provide the current inventory of coatings (Spring 2005 for Unit 2 and Fall 2005 for Unit 1) provide a pictorial representation of degraded or nonconforming coatings at each of the elevations in the containments.

Previous Analyses & Recommendations

In a description of a 1998 analysis performed for Unit 1 (and one at the time pending on Unit 2), the station response to GL 98-04 (Reference 1) states that, "The impact of the pressure drop on the ECCS NPSH margin was then determined and showed that there is sufficient NPSH margin". As discussed in Reference 3, the discoveries of late 2005 regarding erroneous application of the NUREG/CR-6224 head loss correlations revealed the NPSH impact portion of the 1998 analyses to be in error. Re-evaluation of the condition using more recently available test data on the failure modes of coatings (specifically that they would fail to minute particles rather than flakes) reaffirmed the overall conclusion that there would be sufficient NPSH margin.

In addition, the late 2005 and early 2006 evaluations reexamined head losses within the sump outlet valve body, and losses due to the constriction of flow as it entered the valve due to a postulated debris buildup on the screens immediately adjacent to the outlet valve disks. The evaluations concluded that these head losses would also not jeopardize ECCS pump operation as a result of inadequate NPSH. The evaluations also considered the potential for localized flashing due to the reduction in local pressure at the flow constrictions, and found that flashing would be suppressed due to the pressure of the air and non-condensable gases resident in containment.

The details of these evaluations were previously transmitted by Reference 5. The same summary information is included as Enclosure 2 in the responses to Questions 1A and 3.D.4.

Compliance with 10 CFR 50.46(b)(5)

Please refer to Enclosure 2. The responses to Questions 1.A and 3.D.(4) provide a summary description of how the facility, in its current configuration and in light of the concerns documented in Reference 3, complies with 10 CFR 50.46(b)(5).

GL 2004-02 Supplemental Information

A Request for Additional Information (RAI) regarding the response to Event Notification 42129 was transmitted to NMC by the NRC via Reference 4. The NMC commitment to update GL 2004-02 made in Reference 3 has been partially fulfilled via submittal of the NMC response to this RAI on February 16, 2006, identified as Reference 6. Enclosure 2 of that submittal is included as Enclosure 2 of this submittal.

NMC has reviewed the previous GL 2004-02 responses in light of the information developed in late 2005 pertaining to previous analyses of the existing ECCS strainers. NMC has determined that the previous responses pertaining to GL 2004-02 are not materially affected by the issues reported via Reference 3, including the responses pertaining to NPSH, and debris sources (including coatings, insulation, latent debris and miscellaneous debris).

Reference 2 included a description of the design basis debris loading for the screens that were being designed to replace the existing ECCS sump screens. Specifically, the inventory of degraded or non-conforming coatings obtained during previous refueling outages was provided. That inventory was used to establish the working design bases for the replacement screens and it was not intended that the information provided reflect the current inventory. The current inventory was provided in Reference 5 and is again provided as Attachment 1 and 2 of Enclosure 2 of this letter.

NMC recognizes that aspects of the containment coatings program have been inadequate. Root cause evaluation (RCE) 294 was performed to understand the extent of the condition, its causes, and the appropriate corrective actions. RCE 294 determined that the underlying cause of the deficiencies was a failure to describe within the coatings program its safety significance and a failure to establish appropriate acceptance criteria.

Corrective actions to prevent recurrence include establishing the design and license basis limits for degraded and unqualified coatings inside containment consistent with the ongoing efforts to resolve GSI-191, and to update the coatings program documents to reflect the established limits. These actions are being tracked in the PBNP corrective action program.

As work has continued on the design and testing of the replacement screens, the design bases debris loads (including those associated with coatings) have continued to evolve. Programmatic controls associated with Item 2(d)(viii)(f) of GL 2004-02 were previously described in Reference 2, and will be updated as necessary in accordance with the Commission's guidance via letter to Licensees dated March 28, 2006.

ENCLOSURE 2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING EVENT NOTIFICATION 42129 EMERGENCY CORE COOLING SYSTEM (ECCS) LONG-TERM COOLING

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On November 8, 2005, Nuclear Management Company, LLC (NMC, the licensee), notified the U. S. Nuclear Regulatory Commission (NRC) in accordance with 10 CFR 50.72 (Event Notification 42129) that the design basis for long-term cooling at the Point Beach Nuclear Plant (PBNP), Units 1 and 2, was not correctly modeled. NMC's notification stated that, "These errors in the modeling fidelity potentially impact the analytical basis for demonstrating compliance with the acceptance criteria of 10 CFR 50.46(b)(5), Long-term cooling."

On January 10, 2006, the NRC staff issued a Request for Additional Information (RAI) to NMC. The text of the RAI follows as Enclosure 2, with NMC's response to each of the items.

"The NRC staff is reviewing NMC's actions to establish that the requirements of 10 CFR 50.46(b)(5) continue to be met. The staff's review includes the potential blockage of the containment sump and its effect on the ability to sustain long-term cooling, the potential impact of the SI-850 valves to operate so as to sustain long-term cooling, and the potential impact of leakage from the recirculation line, particularly regarding dose to operators. The NRC staff has determined that responses to the following questions are needed to proceed with this review.

For each of the questions below, please ensure that your responses describe your assumptions, methods, and conclusions in sufficient detail to support the NRC staff's independent review. If technical reports are referenced, you should provide a copy of the report and the technical basis for the applicability of the reports to your facility."

1. General

A. Provide a discussion of actions taken to demonstrate the ability to establish and maintain long-term cooling in accordance with 10 CFR 50.46(b)(5).

NMC Response:

There are substantial ongoing reviews that have resulted in proposed changes in the area of containment sump screen design criteria, effects on downstream components, debris sources, debris transport, etc. These changes are being developed and implemented in response to Generic Safety Issue 191 (GSI-191) as communicated to the industry in NRC Bulletin (BL) 2003-01, Generic Letter (GL) 2004-02 and other associated communications. PBNP will comply with the revised acceptance criteria established via GSI-191 by December 2007 as stated in NMC's response to GL 2004-02 dated September 1, 2005.

The following NMC responses are provided within the context of events and discoveries made pertaining to containment coatings and related issues during late 2005. The scope of the responses is confined to the station design and license bases, as they existed at the time of discovery unless otherwise noted.

PBNP has met the requirements of 10 CFR 50.46(b)(5) via the following actions:

<u>1969</u>: The design of the screens predated the issuance of RG 1.82, Revision 0, "Sumps for Emergency Core Cooling and Containment Spray Systems," (issued in 1974), but reflects portions of that guidance. Features include:

- Separation from high energy piping systems by structural barriers (C.2)
- Located on the lowest floor elevation of containment with a trash rack and a fine inner screen (C.3)
- No drains terminating so as to impinge water (and entrained debris) on the screens (C.5)
- A substantial vertically mounted trash rack (C.6)
- A vertically mounted inner screen designed for 0.2 fps with 50% screen blockage (C.7)
- A solid top deck that would be submerged after completion of safety injection (C.8)
- Seismic rack and screens (C.9)
- Screen openings sized based on the minimum restrictions of downstream components (C.10).
- Corrosion resistant materials (C.12)

<u>1989-1990</u>: In response to concerns about blockage of the sump screens by debris, including GL 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," and Regulatory Guide (RG) 1.82, Revision 1, detailed unit-specific analysis of debris generation and transport were performed to consider debris from both coatings and insulation. The potential adverse effects on downstream components and the reactor core were also evaluated. The analyses concluded acceptable performance of the ECCS system without additional modifications or changes to the plant. Copies of these analyses were provided on CD-ROM.

1992: The station developed a hydraulic model of the integrated ECCS system. This model provided the ability to perform more sophisticated analyses and evaluation of alignments and scenarios not previously considered. As a direct result of this model and subsequent refinements to it, the station identified that operation of containment spray in the "piggyback" mode may result in insufficient net positive suction head (NPSH) to the operating residual heat removal (RHR) pump. The option to operate in this alignment was removed from the emergency operating procedures.

1994: Recognized that prolonged simultaneous operation of both trains of emergency core cooling system (ECCS) during the injection phase would rapidly deplete the RWST inventory and challenge the ability to successfully transition to sump recirculation prior to losing the suction source. Emergency operating procedures were revised to direct securing a single train and prolong the suction source. (This change pre-dated the Candidate Operator Actions of BL 2003-01).

1997: Responded to GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," reaffirming adequate NPSH for the ECCS pumps.

1998: Established a refueling frequency inspection of containment coatings and maintaining a detailed inventory of those that are unqualified or degraded. The inspections are reflected in the station's response to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment."

1998-1999: To support the response to GL 98-04, two more unit-specific transport analyses were performed to evaluate the potential for transporting potential coatings debris to the sump screens, and the effect of such blockage on the NPSH available to the emergency core cooling (ECCS) pumps. These analyses used NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage due to LOCA Generated Debris," October 1995, to estimate the potential head loss. The analyses established a "Zone of Influence" that required heightened awareness and maintenance of coatings within the zone.

1999-2005: PBNP maintained an informal inventory of degraded and unqualified coatings inside containment. The as-found deficient coatings were evaluated by informally re-performing the transport and head loss analyses performed in 1998-99 and ensuring that screen head losses were still acceptable.

2003-2005: PBNP implemented Candidate Operator Actions (COAs), as appropriate, and consistent with the NMC response to BL 2003-01. These actions included:

- Operator training on sump clogging.
- Stopping unnecessary redundant pumps in 1994.

2003-2005 (continued)

- Implementing more aggressive foreign material control of containment.
- Ensuring containment drainage paths are unblocked. This was previously performed by installation of a strainer in the refueling cavity drain, and use of reflective metal insulation (RMI) on the reactor vessel head.
- Ensuring sump screens are free of adverse gaps and breaches (Technical Specification surveillance requirement SR 3.5.2.6).
- Initiating analyses necessary to resolve GSI-191 issues.
- Providing a more aggressive cooldown and depressurization following a small break loss of coolant event (LOCA).
- Providing guidance to refill a depleted refueling water storage tank (RWST); providing symptoms and identification of containment sump blockage, and developing contingency plans in response to sump blockage, loss of suction, and cavitation.
- Injecting more than one RWST volume.

2001: NMC determined a potential for a higher head loss across the containment sump outlet valves (1&2SI-850A&B) than previously recognized. This was because the earlier models used the head loss value for a standard valve, yet these outlet valves have an unusual configuration and typical valve head loss factors should not have been used. The head loss was recalculated using a summation of entrance, exit, contraction, and expansion head losses, and the NPSH calculation (N-92-086) was revised accordingly.

2005: NMC evaluated the potential impact of emergent concerns related to the containment sump outlet (1&2SI-850A&B) valve due to the postulated formation of a “debris collar” (see the NMC responses to Question 4 below).

2005: In October/November, it was discovered or recognized that:

- The inventory of degraded and unqualified coatings was no longer bounded by the 1998-1999 analyses.
- The methodology that had been used for calculating the head loss across the screens in those analyses was non-conservative.
- Air entrainment rather than NPSH is the limiting factor for the RHR pumps when operated in the post-accident sump recirculation mode (due to the partially submerged sump screens).
- A postulated “debris collar” around the sump outlet valves could lead to a significantly higher head loss at the sump outlet than previously evaluated.

As a result of these and other findings documented in the corrective action program, several operability determinations (OPRs) and associated corrective action items were initiated. Each of the issues addressed in the operability determinations, the technical basis for the operability determinations, and the conclusions of the determinations are summarized below.

OPR 149, Part 21 Notification of Failed Coatings on Fans: An industry notification of coatings on fans supplied to the nuclear industry was found to be applicable to the replacement control rod drive mechanism (CRDM) cooling fans that had just been installed in Unit 2 (and was operating at full power), as well as, the fans staged for installation with the new Unit 1 reactor vessel head. A knife/pull-off test of the coatings on the Unit 1 replacement fans confirmed that the coatings were deficient. Prior to installation in Unit 1, the vendor removed the deficient coatings and recoated the fans in accordance with the requirements of ANSI N101.4-1972, “Quality Assurance for Protective Coatings Applied to Nuclear Facilities.”

OPR 149 was developed to address the increase in quantity of unqualified coatings present in Unit 2. Revision 0 of this OPR was developed prior to the discovery of deficiencies in the calculations of sump screen head loss due to coating debris blockage (discussed above). The OPR concluded that the incremental increase in overall degraded coatings inventory by the addition of ~160 ft^2 (Revision 1 later estimated the area at 173 ft^2) was minimal and would not impact the operability of the ECCS sump screens.

Following the discovery of deficiencies in the sump screen head loss calculations, the OPR was revised. The revision evaluated the location of the CRDM fans and the potential for transport of fragments of the unqualified coatings to the ECCS sump screens. It concluded that such transport would not occur due to the high density of the fragments, their remote location from the sump, and the low transport velocities that would exist in containment after a design basis LOCA.

The OPR concluded that the sump screens were not challenged by the additional degraded coating inventory, although this condition is a nonconformance to the station license basis commitment that qualified coatings be used for activities comparable in scope and nature to those of the construction phase. Remediation of these degraded coatings is discussed in the response for Question 1.C below. No compensatory actions are associated with this OPR.

OPR 161, Containment Coatings Not Maintained within Analyzed Limits: This evaluation was prompted by the discovery that the total quantity of unqualified and degraded coatings inside the containments was not bounded by the coatings transport and sump screen debris blockage analyses performed in 1998-1999. These analyses had not been recognized as absolute limits, and the analyses had been informally re-performed after each coatings inspection using the increased inventory to check that the acceptable conclusions of the analyses were still valid.

During the development of the OPR, a deficiency was identified in the underlying analyses from 1998-1999. As a result, the head loss portion of the analyses was determined not to be valid.

The deficiency was using the head loss correlation established in

NUREG/CR-6224, “Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris,” to estimate the head loss across the sump screens. The debris bed being analyzed consisted entirely of flat chips or flakes of coatings, and was not postulated to contain any fibrous debris.

However, the correlation published in NUREG/CR-6224 had been established from empirical testing using a mixed debris-type bed consisting of fine particulates and fibers. Use of the NUREG/CR-6224 correlation was, therefore, not appropriate. In the absence of established methods and correlations, there was not a valid, peer-reviewed approach for calculating the head drop across a debris bed consisting entirely of flat plates or chips. The analyses will be superseded by GL 2004-02 analyses and only the portions evaluating the potential transport of coatings debris will remain valid.

To address the immediate concerns of operability, the results of the transport portions of the 1998-1999 analyses were scaled using less limiting, but bounding, values of sump depth and withdrawal rates to determine the critical areas of interest for degraded coatings. In addition, recent testing results, documented in EPRI Technical Report 1011753, “Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings,” September 2005, supported the deletion of unqualified coatings as challenges to the operability of the sump screens. It was concluded that there are not sufficient degraded coatings in proximity to the sump screens to challenge operability of the screens.

This OPR contained a new and conservative assumption equating the area of degraded coatings that could reach the sump screens to the area of the sump screens that would be blocked (e.g., one square foot of degraded coatings equals one square foot of blocked screen surface area). This is conservative because the visual inspections for degraded coatings intentionally round upwardly the areas of degradation observed and because flat platelets (chips, flakes, etc.) would be expected to form a porous debris bed at least a few plates deep, rather than spread out evenly to form an impervious layer one platelet thick. The resulting debris bed would effectively block a considerably smaller screen surface area than the area of degraded coatings that created the debris.

After the initial issuance of this OPR, continuing internal reviews of the coatings inspection results from the prior refueling outages identified a previously unrecognized area of reported degraded coating in close proximity to the Unit 2 containment sump screens. An entry into containment was performed to inspect the area of concern. The reported degradation was confirmed, and a reactor shutdown was commenced in accordance with Technical Specification 3.0.3. During the shutdown, the degraded coatings were reduced to an acceptable level, leaving only coatings that were not accessible without the erection of scaffolding. The shutdown was terminated and the unit was returned to full power operation.

The OPR was revised (Revision 1) to address degraded coatings remaining in proximity to the sump screens. The OPR concluded that these inaccessible remnants were located too far from the sump screens to present a challenge to them (i.e., would not be transportable). Other emergent concerns were also addressed in this revision of the OPR, including other containment latent debris such as tape, labels, and remnants of mineral wool used during the construction of the facility that are still adhering to the bottom surface of overhead floor slabs, and thermal insulation.

Both revisions of the OPR concluded that while the screens are operable, the increased quantity of unqualified or degraded coatings in containment constituted a nonconformance to the license basis as communicated in the station response to GL 98-04. No compensatory measures were indicated, and resolution of the nonconformance will be achieved by completion of the GSI-191 project. That project will supersede the existing criteria for coatings, insulation, etc, and replace them with the design bases assumptions and analyses for the replacement sump screens. For further details of this OPR, please refer to the responses under Question 3 below, OPR 161, Revision 1, and Engineering Evaluation 2005-0024, Revision 1 (provided on CD-ROM).

OPR 162, Ability of Sump Screens to Pass Required Flow: The NRC prompted this evaluation when it was observed that the containment sump screens are in close proximity to the sump outlet valve disc. In the event that a small debris "collar" formed at the base of the sump screens, it would cause outlet flow to be channeled through a narrow annulus between the valve disc and the sump screen. The concern was that the resultant head loss could cause a loss of required NPSH to the RHR pumps.

During development of the OPR, additional concerns (the potential for flashing of hot sump fluid just downstream of the annular constriction, and for air entrainment by vortexing) were also addressed.

The OPR evaluated the potential for both excessive head loss and flashing of the sump fluid. It concluded that head losses would remain acceptably low, provided that the outlet flow rate is limited to that achievable by a non-degraded RHR pump delivering flow to only the reactor core. If the pump were to be aligned to discharge to both the core and a high head SI pump without throttling the total flow, excessive head losses could result. This "piggyback" alignment would occur during operation to flush postulated concentrated boric acid from the vessel outlet plenum many hours after a DBA LOCA. A compensatory measure to limit RHR flow to prevent a loss of NPSH was therefore implemented.

The OPR also concluded that flashing would not occur, but that it was necessary to credit static containment "overpressure" to arrive at this conclusion. The "overpressure" is due to the air present in the post-accident containment. The pressure contribution of this air had been intentionally omitted from previous NPSH analyses as a conservative and bounding assumption. Since crediting this "overpressure" was not consistent with the station's response to GL 97-04[1], this is considered to be a nonconformance with the license basis. Revision 1 of the OPR expanded and clarified the contents of Revision 0 to address additional questions posed by the NRC.

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[1] GL 97-04 was concerned with available NPSH as calculated by the customary two-point method, and did not identify concerns with flashing of hot sump fluid en-route between the sump and the pump impeller. The potential for flashing at some intermediate points, and specifically when passing through a sump screen, was recognized as a result of the new guidance contained in the safety evaluation for NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," dated December 2004.

Continued reviews in response to an inspector question about the potential for the containment sump outlet valves to gradually “drift” shut during long-term containment sump recirculation prompted Revision 2 of the OPR. The revision evaluated how far the outlet valves could drift in the shut direction from fully open until unacceptable frictional head losses and/or flashing would occur. The results established that there would be adequate time to take remedial action, such as reopening the valve, between indications of valve drifting and the loss of the RHR pump suction source.

All three revisions of the OPR concluded that while the ECCS system remained operable, the crediting of containment “overpressure” was a nonconformance to the license basis, and that a compensatory measure to procedurally limit the total sump outlet flow is necessary to ensure that adequate NPSH is available to the RHR pumps. Resolution will be achieved by completion of GSI-191 activities, and may require a license basis change to credit either containment “overpressure” or sump fluid subcooling to demonstrate that flashing at the outlet valve disc will not occur. The current revision of OPR 162 was provided on CD-ROM.

<u>OPR 164</u>, Wax Deposits on Unit 2 Containment Floor. During the Unit 2 at-power containment entry to inspect and remove suspected degraded coatings (discussed in connection with OPR 161 above), areas of dark deposits on the containment floor coatings were observed in the vicinity of the containment sump screens and on an upper elevation of containment. These deposits had been previously documented in containment coatings inspections as remnants of floor “wax”. Previous efforts in the mid-1990s to remove the “wax” deposits throughout containment had been largely successful, however, there were still isolated areas of coatings that had not been removed after repeated attempts.

OPR 164 addressed the presence of these remaining deposits, and established that they were not a wax, but rather, an acrylic co-polymer floor coating. The tenacious nature of the deposits, their limited extent (~40 ft^2), and their benign failure mode (i.e., into fine particulates that would pass through the sump screen perforations) contributed to the conclusion that their presence did not pose a challenge to the operability of the ECCS sump screens.

While the OPR concluded that the Unit 2 sump screens would not be challenged by the presence of the remaining unqualified floor coatings, their existence is considered a non-compliance with the station license basis that will be resolved by completion of GSI-191 activities. No compensatory measures are indicated.

Further removal of the tightly adherent remnants of acrylic floor sealer (“wax”) will not be attempted because previous attempts have resulted in damage to the underlying qualified coatings and concrete. Removal would pose a challenge to quality that is disproportionate to their continued presence. The sizing of the replacement screens is taking into account these unqualified coatings as part of the design basis particulate debris loading.

OPR 170, Design Basis Leakage Detection Capability Defeated. During continuing reviews of the design and license basis for the ECCS sumps and related systems, it was found that an original design feature of the system had been defeated by later actions. In the original design, ECCS leakage originating in piping in the tendon gallery underneath the containment structure would collect in the gallery sump and be channeled through open pipe sleeves to the RHR pump room. This room has instrumented sumps that alarm in the control room in the event of a high level, and this would alert the operator to an abnormal condition, such as excessive leakage in the ECCS system.

Subsequent site activities grouted the pipe sleeves closed. These grouted closures have since been found to be credited as limiting the intrusion of ground water into the RHR pump rooms, providing seismic supports for the RHR piping (in the case of Unit 1), and limiting potential flooding of the RHR pump room in the event of an RWST rupture.

OPR 170 evaluated whether adequate indication of ECCS leakage in the tendon gallery remained despite the closing of the intended drain paths. It concluded that the safety-related containment sump level instrumentation provided ample indication of a loss of sump inventory caused by leakage before it could jeopardize the functioning of the ECCS system. The OPR also considered the potential dose consequences of postulated leakage and found them to be acceptably bounded as well. For further details and information, please refer to the responses to Question 5 below.

OPR 170 concluded that removal of the leakage path from the tendon gallery to the RHR pump room did not jeopardize the operability of ECCS or supporting equipment, but that it did constitute a nonconformance with the design and license basis description of leakage detection capability. Since there are other reliable means of leakage detection (i.e., the redundant and environmentally qualified containment sump level indications in the control room), no compensatory measures were required.

OPR 171, Safety Functions of Containment Accident Sump Isolation Valves: Pursuant to NRC inspection activities and continued internal reviews, it was determined that the containment sump outlet valves have an active function to shut to isolate a postulated system leak occurring downstream of the valves. Since this function had not been explicitly identified previously in station inservice testing (IST) documentation, OPR 171 evaluated whether there was reasonable assurance that this function would be achieved. Further details of the technical issues pertaining to this OPR can be found in the NMC responses to Questions 4 and 5 below.

The OPR concluded that, based upon stroke testing performed incidental to the open stroke testing, refueling frequency leakage testing of the downstream piping, and an initial review of environmental qualification of the supporting components (such as the position limit switches, solenoid pilot valves, and hydraulic power packs), the valves would perform the identified function to shut reliably in the event of a design basis event. However, the condition is nonconforming to the station’s license basis because there is not sufficient environmental qualification documentation for the shut safety function and the testing protocol for this function are not complete.

Quarterly stroke testing procedures for the valves have been revised to verify close-stroke capability. No compensatory measures are necessary. Additional corrective actions are to be taken as discussed in response to Question 1.C below.

In summary, the six OPRs described above concluded that in each case a nonconformance to the license basis existed. However, in each case, the potentially affected systems, structures or components (SSCs) were also determined to be operable.

B. Have you completed a 10 CFR 50.59 evaluation of compensatory measures (e.g., ECCS flow reduction) taken as part of your OPRs? If so, provide a copy of those evaluations. If not, please explain why?

NMC Response:

The only compensatory action directed was to limit the flows through an RHR pump operating on containment sump recirculation to 1560 gpm or less when operating an SI pump in "piggyback."

During safety injection, a single train of RHR discharging through its piping system and against a depressurized RCS has been analyzed to deliver ≤1582 gpm (there are slight variations from train to train and unit to unit due to differences in pipe routing). This is more than adequate for decay heat removal (~200 gpm of boil-off at 20 minutes post-trip), even assuming that 50% of the flow spills to the containment prior to reaching the reactor vessel.

However, when a parallel flow path is aligned from the RHR pump to both the RCS and an SI pump, the decrease in RHR pump discharge back pressure will result in a marked increase in RHR pump flow if no other actions are taken to limit it. The procedural direction to limit the flow results in keeping the RHR pump within the analyzed acceptable condition of 1582 gpm while ensuring sufficient flow for decay heat removal.

Please refer to the response to Question 4.A below for further details. The 10 CFR 50.59 screening of the change to the procedures (SCR 2005-0260) was completed and was provided on CD-ROM.

C. Provide a detailed discussion including planned actions and schedule for resolution of any nonconformances with the current licensing basis or degraded conditions.

NMC Response:

The following actions and schedule for resolution of nonconformances or degraded conditions is provided. The actions and schedule are provided reflect the due dates that are listed and are being tracked to completion in the PBNP corrective action program. At the latest, these actions will be completed consistent with the existing NMC commitment to resolve GSI-191 by December 31, 2007.

Specific Items to be Resolved External to GSI-191

Refueling Frequency Testing of SI-850 Valves: The procedures to stroke test the valves on a refueling frequency will be revised with appropriate acceptance criteria prior to the next performance of each test during each unit's upcoming refueling outage.

Sump Outlet Valve Position Indication Qualification: The position indication limit switches for the SI-850 valves will be dedicated or upgraded to be able to withstand an anticipated harsh environment due to integrated gamma dose prior to the end of the next refueling outage on each unit.

Sump Outlet Valve Motive Power: The hydraulic power packages and positioning solenoid valves for the SI-850 valves will be dedicated or upgraded to be able to withstand an anticipated harsh environment due to integrated gamma dose prior to the end of the next refueling outage on each unit.

Detection of SI System Leakage into the Tendon Gallery: Alternatives to the grouting that currently exists in the tendon gallery are being evaluated. Resolution of tendon gallery grouting issues will be consistent with resolution of GSI-191 and will be completed by the end of the next refueling outage of each unit (Fall 2006 for Unit 2 and Spring 2007 for Unit 1).

Programmatic Guidance for Monitoring Containment Sump Level: Post-accident, long-term programmatic guidance will be implemented by June 2006 to include explicit direction for monitoring the containment accident sump level for adverse trends that may indicate a leak of service water into containment (uncontrolled rise in sump level), or a leak of sump inventory out of containment (uncontrolled drop in containment sump level), and to investigate the condition accordingly.

Remediation of the Unit 2 CRDM Fan Coatings: The non-conforming coatings on the Unit 2 CRDM fans will be removed or the fans replaced with ones that are either uncoated or coated with qualified coatings prior to the end of the fall 2006 refueling outage.

Specific Items of Concern to be Resolved Under GSI-191

NMC continues to pursue resolution of GSI-191 issues in accordance with GL 2004-02 requirements and will provide status updates to the Commission in accordance with the provisions of the GL.

Control of Containment Coatings: The design basis for the replacement sump screens defines the limits of unqualified and degraded coatings that may exist in containment and the location of those coatings. Prior to the end of the next refueling outage on each unit, containment coatings will be removed, repaired, or restored to the extent necessary to be enveloped by this design basis. Subsequent refueling frequency coatings inspections will ensure the total inventory of coatings and other sources of particulate debris will remain bounded.

Sump Screen Replacement: Replacement of the existing sump screens with the GSI-191 replacement screens will eliminate the potential for a “debris collar” flow restriction. Replacement of the sump screens will occur consistent with NMC’s commitment to GL 2004-02, no later than December 2007.

Crediting of Containment Overpressure: Assuming no containment overpressure, there may be a potential for fluid flashing under the sump outlet valve discs, even after installation of the new strainers. However, a minor “overpressure” would suppress such flashing. Substantial overpressure would be available due to trapped air and non-condensibles inside the containment building. Resolution of this issue will occur concurrent with resolution of GSI-191.

2. Zone of Influence

A. What is the zone of influence? How was this determined? What is the basis for this answer?

NMC Response:

Attachments 1 and 2 of Enclosure 2 contain graphical depictions of the Zone of Influence for each containment that are used to assess operability.

The term “Zone of Influence” is defined in the Purpose/Objective section of calculations M-09334-345-RH.1 and M09334-431-RH.1: “The zone of influence is defined as the horizontal distance extending from sump screen projected onto the water surface into which failed coating debris would be transported to the sump screen by the flow of water rather than settling on the containment floor.”

These two calculations were concerned with the potential for failed coatings interacting with the sump screens. The calculations also considered a second Zone of Influence due to particles sliding along containment floor. This extended zone encompasses the area around a screen where coatings debris would settle to the floor, and once on the floor of the containment, could be transported to the screen surface by sliding along it.

While the term “Zone of Influence” was not used in the earlier 1989-1990 unit-specific evaluations of paint and insulation debris effects on containment emergency sump performance, the methodology used to determine the quantity of debris that could be transported to and accumulate on the debris screens was comparable. The result was a graphical depiction of a “Debris Transport Zone” in Figure 6.2-3 of the evaluations.

A term with the equivalent meaning of the “Zone of Influence” (as historically used) is “Zone of Transport”. “Zone of Transport” denotes the region surrounding the sump screens where suspended debris would ultimately arrive at the surface of the screen by all modes of transport combined. This response will state “ZOI/ZOT” when describing this region of potential debris transport. Upon resolution of issues related to GSI-191, the previous analyses will be obsolete and the terms “Zone of Influence” and “Zone of Transport” will be used consistent with their use in NEI 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” and its associated NRC Safety Evaluation.

<u>How the ZOI/ZOT was Determined</u> - The method for determining the ZOI/ZOT in calculations M-09334-345-RH.1 and M09334-431-RH.1 is described in the Methodology/Acceptance Criteria sections of those documents. The methodology used for calculating the horizontal water velocities and coating transport mechanisms is based on NUREG/CR-2791, “Methodology for Evaluation of Insulation Debris Effects,” September 1982.

Conceptually, the settling velocity of a postulated coating fragment is determined using the coating density, the assumed characteristic dimensions of the coating fragment (establishing the drag coefficients), and the density and viscosity of the liquid that it is sinking through. The time it takes the postulated fragment to sink through a sump of given depth is then determined. In the subject calculations, both the minimum and maximum sump depths were used to ensure bounding results were obtained. Using hydraulic flow modeling methods, the flow field velocities for the areas surrounding the containment sump screens were determined. Multiplying this radial flow velocity approaching the sump screens by the settling time for a debris fragment in a given depth of water results in the characteristic length of the ZOI/ZOT for direct impact on the sump screen surface.

The extended ZOI/ZOT that includes transport by sliding along the containment floor is a two-step process. In the extended ZOI/ZOT, the minimum bulk velocity to cause sliding of a postulated coating fragment is first determined, and the region surrounding the screens with flows at or above this velocity is then established. Particles reaching the floor within this region may be expected to transport to the base of the sump screens.

After this sliding region has been established, the process of determining horizontal transport during settling, the same as was done for the direct screen impact, is repeated. Adding the two characteristic lengths (one for sliding transport and one for settling transport) results in the final characteristic length of the ZOI/ZOT for sliding transport.

The calculation of the ZOI/ZOT was refined, where appropriate, to differentiate between flows originating from different areas surrounding the sump screens. This was because the calculated horizontal velocities varied depending upon the flow channels being considered. The results are illustrated in Figures 9, 10, and 12 of calculation M-09334-345-RH.1, and Figures 7, 8, and 9 of calculation M-09334-431-RH.1.

<u>Recent Revisions to the ZOI/ZOT</u>: In late 2005, the ZOI/ZOTs contained in the previous evaluations were re-reviewed in Engineering Evaluation 2005-0024. It was recognized that the earlier evaluations had assumed lower water levels and higher flow rates than would exist under the current operating procedures and equipment limitations. By taking a ratio of the maximum supportable sump flow rate to the flow rate assumed in the earlier evaluations, the size of the direct impact ZOI/ZOT was reduced accordingly. The details of the reduction are presented in Engineering Evaluation 2005-0024, Revision 1, which was provided on CD-ROM. That evaluation determined that the largest horizontal projection of the ZOI/ZOT in either unit is bounded by a maximum of 2.4’ based on a flow channel in Unit 2.

The worst-case ZOI/ZOT dimension for direct embedment on the screen surface determined in calculations M-09334-345-RH.1 and M-09334-431-RH.1 was 7.3’. This had been calculated to exist in Unit 2 at a minimum flood level of 2.68’ and a flow rate of 4,847 gpm. The corresponding calculated ZOI/ZOT for a maximum flood level of 6.18’ was only 6.6’. This demonstrates a diminishing ZOI/ZOT size for this flow channel with increasing flood depth. After scaling to account for the actual expected lower flow rates (reflecting equipment limitations and the use of only a single train during sump recirculation), the characteristic size of this worst-case (Unit 2) ZOI/ZOT was reduced to 2.4’.

Based on recently completed test results contained in EPRI Technical Report 1011753, “Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings,” September 2005, the ZOI/ZOT for sliding transport was eliminated. ZOI/ZOT had been based on postulated low-density alkyd coatings specific gravity of 1.12, and that had been assumed to deteriorate to transportable chips, which could then block the screen perforations.

The recently completed EPRI testing demonstrated that disintegration products from such coatings would be small particulates that are not capable of lodging in the screen perforations. The effects of ingestion of these small particulates are addressed in the NMC response to Question 3.E below.

In summary, the current ZOI/ZOT of concern is based on the settling velocity of qualified (acceptable) epoxy coatings only. This ZOI/ZOT is based on impingement of debris on the screen surface prior to settling on the sump floor. It also includes all locations where transport by other credible mechanisms could result in the deposition of the coatings fragments at the surface of the water within the ZOI/ZOT. Examples are degraded epoxy coatings which are located on the containment liner plate directly above the containment sump ZOI/ZOT or containment spray water wash-down of the vertical liner plate in this region could result in the fragments being carried to the sump area adjacent to the screens. The worst-case characteristic size for the ZOI/ZOT is 2.4' from the sump screen surface. With additional refinement, this dimension could be further reduced by considering unit and scenario-specific parameters. The NMC response to Questions 3.A and 3.D(2) discuss the coatings in this ZOI/ZOT.

Attachments 1 and 2 of this enclosure contain maps of containment depicting the ZOI/ZOT on the elevations in the PBNP reactor containments. The ZOI/ZOT depicted on El. 8' is limited to the area immediately surrounding the screens as discussed above. In addition, it has been the practice to include an arc of the containment liner adjacent to the sump screens and extending all the way to the containment dome as also being within the ZOI/ZOT. Piping and components in proximity to this arc have also been considered within the ZOI/ZOT unless otherwise evaluated. These inclusions were based on postulated wash-down of degraded or unqualified coatings in these areas reaching the screen surface during containment spray operation. This was applicable when the ZOI/ZOT was large enough to extend to the containment liner wall. Although the ZOI/ZOT has contracted (as discussed above), the vertical extensions of the ZOI/ZOT are retained due to the turbulence of the pool adjacent to the liner caused by the sheeting and cascading of water coming down the vertical liner plate during containment spray operation. The arcs associated with the vertical extensions of the ZOI/ZOTs are also depicted at each elevation of the containment on the maps provided.

3. Potential Blockage of the Sump/Long-term Cooling

A. Containment Coatings

(1) How much (percentage, area, and volume) of the coatings will fail? Include the location of the failed coatings, the type of coating, and qualification level of the coatings. What is the basis for this answer?

<u>NMC Response</u>:

There are two general types of coatings that are assumed to fail and be released to containment during or after a design basis loss of coolant (LOCA) event. These are unqualified coatings, and coatings that are qualified (Acceptable) but have become degraded by means of de-bonding or delaminating. The following tables summarize the quantity of each type located in each of the two containments based on inspections performed during the last refueling outage on each unit.

Unit 1	Total Area (ft^2)	Percent of Total Coatings Area (%)	Total Volume (ft^3)
Unqualified Coatings	19,747	5.6	3.5
Acceptable but De-bonding/ Delaminating Coatings*	996	0.028	1.6

Unit 2	Total Area (ft^2)	Percent of Total Coatings Area (%)	Total Volume (ft^3)
Unqualified Coatings	21,826	6.2	3.9
Acceptable but De-bonding/ Delaminating Coatings*	3,940*	1.1	6.2

*An additional ≈173 ft^2 of degraded coatings were subsequently identified as a result of a 10 CFR 21 notification. That notification dealt with improperly applied coatings on the recently replaced control rod drive mechanism (CRDM) fan housings. It was determined that this additional inventory was insignificant and was located outside of a ZOI/ZZOT of concern. The fans with deficient coatings designated for installation in Unit 1 were replaced with fans that had fully qualified coatings prior to actual installation.

During the most recent refueling outage (Unit 1), the coatings inspection differentiated between de-bonding/delaminating coatings and those that were degraded in other benign modes, such as mechanical abrasion or impact damage, cracking but tightly adherent, etc. This distinction had not been previously applied, and results in the Unit 2 inventory being substantially larger. The quantity of degraded coatings in Unit 2 is inferred from the textual descriptions contained in the inspection reports and is believed to be conservative because it does not differentiate between types of degradation

Unqualified coatings are widely distributed throughout the containments in relatively small quantities, but the main sources are attributable to a few discrete components: (1) Polar crane (~5,500 ft^2); (2) Polar crane rail girder (~4,950 ft^2); (3) Manipulator crane (1,500 ft^2); (4) Reactor coolant pump motors (600 ft^2).

Attachments 1 and 2 of this enclosure provide a detailed listing of the delaminating “Acceptable” coatings in each containment, followed by graphical depictions of their approximate locations.

Acceptable coatings are coatings that include coating systems which have been reviewed for suitability for application inside containments, and there is reasonable assurance that the coatings will not detach under normal or accident conditions. At PBNP, the coating systems specified for use on major structures during original construction were tested and qualified for the design basis accident (DBA) environment by WCAP-7198-L, “Topical Report – Evaluation of Protective Coatings for Use in Reactor Containment,” dated April 1, 1968.

Unqualified coatings are those coatings do not meet the above criteria. These are mostly original equipment manufacturer (OEM) applied alkyd (oil) based coatings. A coating lacking sufficient documentation to establish it as a “Qualified” (Acceptable) coating is classified as unqualified.

Unqualified coatings are assumed to all be alkyd-based and 100% of them are assumed to fail. As discussed in the NMC response to Question 3.A.(2) below, the failure products of these coatings are benign, do not challenge the functioning of the ECCS sump screens, and are not represented by a detailed listing or graphical depiction of location. Additionally, only qualified (Acceptable) coatings that exhibit delamination or de-bonding are assumed to fail.

The total coverage of coatings is approximately 353,100 ft^2 per containment. This value was used as the basis for determining area percentages of failed coatings. The total volume of coatings is determined by multiplying the thickness by the area (see the NMC response to Question 3.A.(2) below).

(2) What are the physical characteristics of the failed coatings (particle size, thickness, and specific gravity)? What is the basis for this answer?

<u>NMC Response</u>:

<u>Size</u>: Unqualified coatings are assumed to fail to minute particles bounded by 1128 microns or less in characteristic dimension. This is based on EPRI Technical Report 1011753, “Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings.” This recently issued report demonstrates that a broad range of coatings, including epoxies and alkyds, when they deteriorate, do so in the form of fine particulates.

Acceptable coatings that fail (de-bond and become available for transport) are assumed to be flat discs 1/8” in diameter. This assumption is based on having the smallest possible fragment that could physically lodge in or on the 1/8” screen perforations. By minimizing the size the transportability of the fragments is maximized.

The flat disc also maximizes the drag coefficient such that both the settling velocity and the velocity of water necessary to transport horizontally across a surface are minimized.

Thickness: The thickness of coatings varies by application and location. The values used in various analyses depend upon the purpose of the analysis (i.e., whether it is evaluating the heat transfer to containment heat sinks to calculate the pressure and temperature response to a LOCA, whether it is evaluating the quantity of debris that may be generated, etc.). A sampling of existing coatings thickness was used to establish a conservative value for the debris generation analyses of interest.

The Dry Film Thickness (DFT) of unqualified alkyd coatings was measured to be between 0.0003 and 0.0038”, with an average of 0.00212”. This value is appropriate when estimating the total volume of such coatings. The DFT of acceptable (epoxy) coatings was measured between 0.0045” and 0.0187” with an average of 0.0116”. When evaluating the transportability of these coatings, a conservatively low value is appropriate for determining transportability (0.005” was used in most cases, although 0.015” was used where justified for the concrete floor coatings in Unit 1).

Based on the above, for the purposes of estimating the total volume of epoxy coatings, a bounding high value of 0.019” was used for dry film thickness.

Specific Gravity: The specific gravity of unqualified (alkyd) coatings used in the previous transport analysis was 1.12. However, this value is not relevant since PBNP is assuming that these coatings fail to fine particulates and are highly transportable.

The specific gravity of the acceptable (qualified) coatings used at PBNP is bounded by a low value of 1.6. This reflects the specific gravity of the Phenoline 305 coatings used on concrete surfaces inside containment (~85,000 ft^2 per containment). The other acceptable coating systems consist of Dimetcote 6 primer (specific gravity 3.2), and Amercote 66 (specific gravity of 2.6). These two higher density coatings were used on the major steel surfaces of containment such as the containment liner and structural steel (~268,000 ft^2 per containment).

Summary Of Failed Coating Particle Characteristics

Coating Type	Failed Particle Characteristic		
	Size	Thickness	Specific Gravity
Qualified / Acceptable (Epoxy)	1/8” diameter discs	0.019”	1.6 minimum
Unqualified (Alkyd)	<1128 microns	0.00212”	1.12

(3) **Will the failed coatings be transported, including during the blow down phase of the event, to the sump? What is the basis for this answer?**

NMC Response:

Unqualified Coatings: The disintegration products of unqualified coatings are conservatively assumed to be 100% transportable to the sump owing to their minute sizes.

Acceptable but Degraded (Delaminating) Coatings: Acceptable coatings that fail into chips or flakes large enough to be a challenge to the screens are also too dense to be readily transported by the low velocity flows that would exist during sump recirculation. As a result, provided delaminating coatings are located outside of the small ZOI/ZOT, they would not be transported to the sump screens. The deep, flat-bottomed pool with relatively wide, open flow channels, and a low withdrawal rate leads to very low flow velocities (less than 0.1 fps) that are not conducive to the transport of negatively buoyant debris. As a result, Acceptable coatings debris large enough to pose a challenge to the ECCS sump screens will not transport to the screens.

Supporting Details: The PBNP "sumps" are not conducive to the transport of debris. The sumps are not depressed sumps, but rather, comprise the entire El. 8' of the containment. This floor is nominally flat with the sump outlet pipes dropping vertically downward from the floor level. The opening of the 10" outlet pipes (one per train) is flush with the floor. As can be seen in Attachments 1 and 2 of this enclosure, the floor plan of El. 8' of containment is relatively open and unobstructed. This minimizes high velocity channels and choke points, and therefore, minimizes re-suspension of settled debris.

The absence of a depressed sump precludes a "trap" that could collect debris during the energetic blowdown phase of a transient. Switchover to the containment sump is directed when indicated refueling water storage tank (RWST) level is 34% or less. This corresponds to an actual level in the containment sump of ~42" above the containment floor. This figure discounts a contribution from the breached reactor coolant system (RCS) and the safety injection (SI) accumulators.

After initial switchover to recirculation, the containment sump continues to fill as the RWST is depleted using the containment spray system. The final level when spray is terminated at 9% indicated RWST level, the depth of water in the containment sump is ~60".

During containment sump recirculation, only a single train is placed in operation. The hydraulic analysis of the SI and RHR system shows that system piping friction limits total flow to ≤1,582 gpm in this mode of operation. Later, if concurrent upper plenum injection or "core deluge" (the nominal recirculation flow path supplied by an RHR pump) and cold leg injection (supplied by an SI pump operated in series with an RHR pump) is desired for prevention of boron concentration and precipitation, the total flow is procedurally limited to 1,560 gpm indicated flow.

How much (percentage, volume, particle size) of the coatings will be transported including during the blow down phase of the event? What is the basis for the answer?

NMC Response:

Transport during Blowdown: Degraded qualified coatings that fail by delamination are most likely to do so as flakes or chips during the blowdown phase of a postulated transient. However, in that case, they would come to rest on the containment floor before sump recirculation is initiated, would be sequestered, and not available for transport to the surface of the sump screens when sump recirculation is initiated. This is because the horizontal fluid velocities during sump recirculation (less than 0.1 fps) are less than that needed to transport settled debris (0.2 fps). This would be true regardless what mechanism may be postulated to generate the coatings fragments.

As a result, transport of coating chips or flakes to the screens is not considered during the blowdown phase of the transient. The current approach reserves the full inventory of degraded (delaminating) coatings for the recirculation phase. During this phase, a moving fluid field exists that could transport the coatings debris to the screen while they were sinking if the debris landed in close proximity to the screen. The assumption of no transport during the blowdown phase is more conservative than assuming otherwise.

Transport of fines from erosion of qualified coatings and disintegration of unqualified coatings is stipulated. These, however, are incapable of producing sump screen blockage due to their small size. These fines are therefore not considered analytically when evaluating sump screen blockage.

Transport During Recirculation - Acceptable ("Qualified") Coatings: The 1/8" flakes or chips of acceptable ("qualified") coatings discussed in the NMC response to Question 3.A.(2) above are too dense to be transported horizontally across the nominally level floor of containment by the low velocity prevailing flows (less than 0.1 fps) during sump recirculation. Therefore, unless degraded coating fragments are deposited on the surface of the pool at or within the ZOI/ZOT (as described in the NMC response to Question 2.A above), they will not be transported to the sump screens during sump recirculation.

Quantity of Acceptable Coatings that Transport to the Sump Screens

	Unit 1	Unit 2
Area of coverage (ft^2)	0	0
Percentage of all coatings	0%	0%

Volume (ft^3)	0	0
Particulate Size (inches)	0.125	0.125

Transport During Recirculation - Unqualified Coatings: It is assumed that 100% of the unqualified (alkyd) coatings will be transported to the sump area. This is because the fine particulate nature of the disintegrated coatings renders them highly transportable.

The area of unqualified coatings in containment was tabulated in the NMC response to Question 3.A.(1) above and is provided below with the percentages and volumes that they represent. The range in particle size is from EPRI Technical Report 1011753, "Design Basis Accident Testing of Pressurized Water Reactor Unqualified Original Equipment Manufacturer Coatings."

Quantity of Unqualified Coatings that Transport to the Sump Screens

	Unit 1	Unit 2
Area of coverage (ft^2)	19,747	21,826
Percentage of all coatings	5.6%	6.2%
Volume (ft^3)	3.5	3.9
Particulate Size (microns)	5 - 1128	5 - 1128

How much of the degraded qualified and unqualified coatings are on the containment floor (both pre-existing and event generated) in the zone of influence around the sump, and how much of those will be transported to the sump? What is the basis for this answer?

NMC Response:

Acceptable (but Degraded) Coatings: Walkdowns of the floor areas on El. 8’ of containment determined that large portions have some extent of mechanical damage such as abrasions or “dings.” There was no delaminating noted. Therefore, no pre-existing debris from otherwise acceptable coatings is anticipated on the floor in the area immediately adjacent to the sump screens. As discussed in the NMC response to Question 2.A above, the ZOI/ZOT at PBNP is relevant only for coating debris that may be settling through a moving water column. This is because it was determined that coatings debris large enough to pose a challenge to the sump screens is not transportable across the floor of containment.

The maximum flow velocities on containment El 8’ (the entire “sump”) are less than 0.1 fps. This is below the 0.2 fps threshold necessary to transport debris across the floor. As flow converges near a containment sump screen, it will accelerate due to the decreasing flow area normal to the direction of flow. Using a minimal sump depth of 3.2’ (from Calculation 2000-0044, Revision 3; assumes a minimum RWST draw-down, no contribution from spilled RCS inventory, no contribution from SI accumulators, and no expansion due to thermal heating of the sump contents), and a flow rate of 1600 gpm (3.56 ft^3/sec) flowing toward the cylindrical screen, the 0.2 fps threshold would be a cylinder of 1.8’ in diameter. This is smaller than the minor dimension of the trash rack covering the screens (as seen in Figure 4.1-2 of the Gibbs & Hill reports, the trash rack covering the screens is 2’ wide and 5’ long). Therefore, particles large enough to lodge on the screen surface and originating outside of the trash rack are not subject to transport to the sump screens, even at minimum sump depth.

Since there is no ZOI/ZOT for horizontal transport of acceptable coatings debris large enough to challenge the sump screens, there are no acceptable coatings located on the floor within the ZOI/ZOT.

Unqualified Coatings: Unqualified coatings are expected to disintegrate into fines that would not pose a challenge to the functioning of the ECCS screens. They are, however, assumed to be 100% transportable to (and through) the sump screens. The total quantity of potential debris was provided in the NMC response to Question 3.A.(1) above. See the response to Question 3.B below for the basis of not having fibrous debris loading on the screens (no thin bed effect).

4) What percentage of the sump screen will be blocked by failed coatings or by coatings in combination with other material? What is the basis for this answer?

NMC Response:

Since the only coatings that are transportable to the sump screens are those that are smaller than the screen perforation size, no blockage of the sump screens due to coatings is anticipated (0% blockage per analysis).

B. Containment Insulation:

(1) How much (percentage, volume, type and size) of the insulation will fail, including during the blow down phase of the event? What is the basis for this answer?

NMC Response: The following tables summarize the quantities of insulation debris generated from the limiting break locations in each unit:

Unit 1 Insulation Debris

Insulation Type	Area or Volume of Debris
Reflective Metallic foils (ft^2)	19,438
Asbestos & Calcium Silicate* (ft^3)	222
Encapsulated Fiberglass (ft^3)	95
Temp-Mat Blankets (ft^3)	67
Encapsulated Mineral Wool (ft^3)	12

Unit 2 Insulation Debris

Insulation Type	Area or Volume of Debris
Reflective Metallic foils (ft^2)	8,862
Asbestos & Calcium Silicate* (ft^3)	301
Encapsulated Fiberglass (ft^3)	95
Temp-Mat Blankets (ft^3)	67
Encapsulated Mineral Wool (ft^3)	12

*NRC Information Notices IN 2005-26 and IN 2005-26a communicated a concern with the presence of calcium silicate (CalSil) insulation in containments that use tri-sodium phosphate (TSP) as a pH buffer. TSP is not used at PBNP. The concern is that the relatively insoluble compound of calcium phosphate will precipitate if there is an appreciable quantity of dissolved Ca^{+2} and PO_4^{-3} ions present in the post-accident solution. It has been postulated that CalSil, while a relatively inert covalent compound, could still contribute significant concentrations of Ca^{+2} into a phosphate-rich sump from the resulting in-clogging (or "blinding") of a pre-existent fibrous debris bed. NMC is aware of these concerns, and has been participating in industry efforts to further quantify these and other potential "chemical effects."

A TSP buffer is not used at PBNP. At PBNP, a sodium hydroxide additive (NaOH) to the containment spray buffers the sump pH. To date, sodium hydroxide buffers have exhibited some potential for the formation of sodium aluminum silicate and aluminum oxyhydroxide (AlOOH) precipitates. This research is being incorporated in the GSI-191 project, as applicable.

In 1989-1990, prior to the debris generation analyses performed in support of the continuing GSI-191 resolution effort, Gibbs & Hill performed debris generation and transport analyses for PBNP (provided on CD-ROM). These analyses followed the general methodology of NUREG/CR-2791, NUREG/CR-3616, and NUREG-0897, Revision 1. The analyses form the current design bases for insulation debris transport.

The analyses evaluated the generation of debris from five categories of insulation installed in the PBNP containments and on or in close proximity to the RCS piping:

- Reflective Metallic
- Asbestos and Calcium Silicate Blocks (with stainless steel jackets)
- Encapsulated Fiberglass
- Temp-mat Blankets
- Encapsulated Mineral Wool

The evaluated mechanisms for debris generation were:

- Jet Impingement (7-pipe diameter zone of destruction)
- Pipe Whip (all insulation between the break and the plastic hinge)
- Pipe Impact (5 fabricated lengths of installed insulation on the impacted pipe)

Although the PBNP licensing basis does not require the consideration of the dynamic effects of a LOCA (modified GDC-4 per 10 CFR 50 Appendix A GDC-4), pipe whip and pipe impact were included in the evaluations.

The limiting break in each containment was determined to be in the “B” steam generator cubicle because of its proximity to the sump screens. A hot leg break was found to be the worst case.

(2) What are the physical characteristics of the failed insulation (particle size, thickness, and specific gravity)? What is the basis for this answer?

<u>NMC Response</u>:

The characterization and evaluation of debris from failed insulation was performed in the Gibbs & Hill reports (provided on CD-ROM). The following are excerpts from Section 7.4.3 of those analyses. Owing to the non-transportability of most of the debris considered, a more detailed characterization of the failed insulation was not performed.

- ”Reflective Metallic Insulation (From Alden Research Laboratory test data reported in NUREG-0897 Revision 1 and NUREG/CR-3616) …”
- “Single sheets of thin stainless steel materials (such as the 0.00025” – 0.004” thick foils used within reflective metallic insulation units)…”
- “As fabricated reflective metallic insulation units…”
- “Outer covers (0.037” thick)…”
- “Inner covers…” (no thickness cited, but apparently comparable to the outer covers)
- “End covers…” (no thickness cited, but apparently comparable to the other covers)

Asbestos, Mineral Wool, and Calcium Silicate Blocks

- “…hard, cast material like mortar with a minimum specific gravity which is greater than water. This material is covered with stainless steel jacketing. If the jacketing is destroyed by jet forces and the block material is also damaged, this material will break into large chunks and fall to the floor…”

Encapsulated Fiberglass and Temp-Mat Blankets:

- “...Type “E” glass… [in] jacketing”
- “…[intact] dislodged panels…” as well as “loose insulation”:
- “…type “E” glass… density… 11 lb/cu ft….”
- “Unlike conventional fiberglass, Type “E” glass is a woven material, not readily subject to ripping and shredding…not anticipated that the Type “E” glass material would disintegrate in such a manner as to allow transportation of glass fibers to the sump screens.”
- “…Absorbs water, particularly hot water, and sinks rapidly (from 20 seconds to 30 seconds in 120°F water)…”

Encapsulated Mineral Wool:

- “…encapsulated in welded stainless steel jackets” or “…encapsulated in welded and riveted stainless steel jackets”
- “In the event of a pipe break… highly unlikely [to be] removed from the jacketing…”

- “Although intact mineral wool mats could be lighter than water, the fragmented fibers have a specific gravity greater than 2…”

(3) Will the failed insulation be transported, including during the blow down phase of the event, to the sump? What is the basis for this answer?

NMC Response:

No significant quantity of failed insulation is expected to be transported to the sump screens, including during the blowdown phase of an event.

Blowdown Transport: Transportation of debris during the blowdown phase of a LOCA event is acknowledged. This mode of transport has not been analyzed in detail, except within the context of the continuing effort to resolve GSI-191 concerns.

The chaotic relocation of such debris during blowdown would tend to be a general dispersal away from the break location, but would not tend to deposit debris preferentially upon the trash rack nor fine screens located within it (for a depiction of the sump, trash rack, and screen configurations, please refer to Figures 4.1-1 through 4.1-3 of the Gibbs & Hill reports included on the provided CD-ROM). Since the “sump” is the entire El. 8’ of containment, there would be no tendency to trap and retain transitory debris passing through the vicinity of the sump screens as could be the case for screens located in a depressed sump.

Further, the debris would subsequently be covered by the rising water level in the containment, be washed down into the deepening pool by continued spray or break flow (and subsequently sink), or remain where deposited on higher elevations. In any case, they would be sequestered and would not be available for further transport once the recirculation flow was initiated.

The current design basis analyses assumed a deposition of debris generated on the floor of the lowest level beneath the loop compartments. This concentrated the maximum quantity of debris in the pool at a location close to the sump screens (in this case, the limiting B loop rupture discussed in the response to Question 3.B.1 above). No deduction was taken for debris blown up to the higher elevations of containment or held up on the bar grate platforms underlying most of the RCS loop compartments.

Subsequent Transport: During the time period between the blowdown transient and the initiation of sump recirculation (while the containment sump fills), there is sufficient time for all debris generated to become thoroughly wetted and sink to the bottom of the containment sump. Subsequent transport to the sump screens would require horizontal transport, and the flow field necessary to cause such transport could not exist until sump recirculation is initiated.

The minimum velocity required to transport submerged insulation debris is 0.2 ft/sec as established in NUREG-0897, Revision 1. The drag force of a submerged object in a freely moving fluid is proportional to the square of the velocity, and the velocity in containment is less than 0.1 fps (see the response to Question 3.A.3 above). Since this is less than half of the empirically observed threshold for transporting sunken objects, there is a drag force margin of at least four (4) between the forces that could exist under post-DBA recirculation and the force necessary to transport the postulated debris. The margin is even higher when it is recognized that the fluid along the floor of the containment is not a freely flowing fluid, but rather has a significant stagnant or slower moving boundary layer that will tend to trap fines and fibers small enough to be fully enveloped in it.

Based on the above considerations, none of the insulation debris that may be generated is expected to be transported to the existing sump screens.

How much (percentage, volume, particle size) of the insulation will be transported, including during the blow down phase of the event? What is the basis for this answer?

<u>NMC Response:</u>

As discussed in the previous response, no insulation is expected to be transported to the sump screens, including during the blow down phase of an event.

(4) What percentage of the sump screen will be blocked by failed insulation or by insulation in combination with other material? What is the basis for this answer?

<u>NMC Response</u>:

No blockage of the screens is anticipated. As was discussed in response to Question 3.A above, and will be discussed in the responses to Questions 3.C and 3.D below, analytical treatment consistent with the current license bases of debris other than insulation, found none that are transportable to the containment sump screens. Therefore, no aggregate effect is indicated.

In addition, since none of the insulation debris is transportable to the sump screens, no blockage of the sump screens due to insulation is expected.

C. **Containment Debris:**

(1) **How much (volume, type and size) containment debris will be transported to the sump? What will happen during the blowdown phase? What is the basis for this answer?**

NMC Response:

No containment debris is expected to be transported to the sump, including during the blow down phase of a postulated transient. The approach taken to determine transportability of containment debris is the same as that for debris originating from coatings and insulation. The debris sources specifically evaluated are tape and adhesive labels known to reside or suspected to remain in small quantities in the containment buildings. This type of debris would pose the greatest potential of both transport (due to relatively low density and high surface area) and screen blockage (due to potential for blocking a significant fraction of the screen surface with an impervious membrane).

The tape widely used for various purposes during refueling outages is a 2" wide fabric reinforced tape commonly referred to as "Duct Tape". Common experience indicates that the adhesive of this tape is thermoplastic, and remnants that may be inadvertently left in containment after an outage cannot be expected to remain adherent under accident conditions. Additionally, an undetermined quantity of conduit marking labels and striped reflective tape remain in each of the containments. Though not tested, it is expected that the adhesive on these items would similarly fail under accident conditions.

The specific gravity of samples of these tapes and labels were measured under ambient conditions, resulting in a specific gravity measurement referenced to room temperature water. However, the density of sump water early in an accident sequence would be lower and the specific gravity of the debris correspondingly higher. This conservatively maximized the analyzed potential for horizontal transport. This is because the frictional forces on debris from contact with the containment floor (those that tend to retard or prevent flow-induced transport) are proportional to the negative buoyancy of the debris. The average specific gravities measured ranged from a low of 1.1 to a high of 1.3.

Since the potential debris sources tested have a specific gravity greater than 1.05, they are not expected to be subject to horizontal transport across the floor of containment with the analyzed flow velocities of less than 0.1 fps. This is based on the guidance provided in RG 1.82, Revision 1 that indicates a velocity of 0.2 fps or greater is required to transport debris of this density.

Transportation of debris during a postulated blowdown event would be inevitable. The distribution of transported debris would be expected to be a general dispersal outward from the break location.

This condition has not been analyzed in detail prior to the development of analyses supporting the continuing effort to resolve GSI-191 concerns.

The evaluation performed to assess current operability (Engineering Evaluation 2005-0024, Revision 1 and OPR 161, Revision 1 assumed a non-specific deposition of debris on the floor of the lowest level of containment. This is consistent with the design basis analyses previously performed for other debris types that are provided in Enclosure 3. No deduction was taken for debris blown up to the higher elevations of containment, or debris sequestered at other locations in containment.

During a postulated blowdown transient, labels and tape that may reside in the containment could be relocated. The chaotic relocation of such debris during blow down would tend to be a general dispersal away from the break location, and would not tend to deposit debris preferentially upon the trash rack, much less the fine screens located within it. Since the "sump" is the entire El. 8' of containment, there would be no tendency to trap and retain transitory debris passing through the vicinity of the sump screens as could be the case for screens located in a depressed sump.

Further, the debris would subsequently be covered by the rising water level in the containment, be washed down into the deepening pool by continued spray or break flow (and subsequently sink), or remain where deposited on higher elevations. Thus, the debris would be sequestered and would not be available for further transport once the recirculation flow was initiated.

Other debris types that could be postulated in the category of "containment debris" are latent dust and dirt, "tramp" (loose individual) fibers, and particles resulting from the erosion of concrete during the blow down phase. Consideration of these debris types is currently outside of the PBNP licensing bases, but they are being included in the analyses necessary to resolve issues related to GSI-191. In the interim, the above evaluations of insulation and coatings debris transport provide reasonable assurance that the probability of transport of such postulated debris is very low. The same reasoning and evaluations methods used in considering those debris types are applied to miscellaneous fines below:

Dust, dirt, and concrete erosion products would, by their nature, either be very fine and capable of passing through the screens unimpeded, or if sufficiently large, would be of a density too high to be transportable. <u>Mark's Handbook for Mechanical Engineers</u> tabulates typical specific gravities for concrete (2.2-2.4), dry sand and packed gravel (1.6-1.9), and damp clay (1.7). These are considerably greater than the 1.05 minimum threshold for transport in a 0.2 fps fluid field established in response to Question 3.a above. This indicates that such particles will sink, and will not transport if already on the floor of containment at the time recirculation initiates.

Further, published studies of the transport of solids by moving fluids have demonstrated that fines transportable at velocities of 0.2 fps and lower are on the order of 1 mm (0.04") or less in size. As such, they would be too small to lodge on the strainer surface, and would pass unimpeded through the 1/8" perforations.

Loose clumps or individual fibers could be postulated to originate from fibrous insulation and would be expected to behave as described in the response to Question 3.B above. This type of insulation debris would be sequestered on the floor of containment after having been wetted out. The source of this type of debris could be from clothing worn in containment (in which case it would be trace amounts whose effects would be too small to quantify), or from filter material residing in containment.

The only filter materials in the various containment ventilation systems are enclosed in plenums of the containment cleanup system. The plenums are located on or above the El. 66' refueling floor, and are located away from LOCA zones of destruction. As a result, this material (or loose fibers originating from it) is not subject to transport during either the blow down or wash-down phases of a postulated event.

Filters that may be brought in to support outage activities, such as high efficiency particulate filters, are required to be removed prior to returning the unit to operation. This is assured by the containment closeout inspection, which requires inspectors to enter accessible containment areas and to ensure the area is free of tools, equipment, dirt or debris accumulation and/or materials that could inhibit Sump "B" recirculation.

Based on the above considerations, transport of containment debris to the sump screens is not expected.

(2) What are the physical characteristics of the debris (size, shape, thickness, and specific gravity)? What is the basis for this answer?

<u>NMC Response</u>:

The physical characteristics of possible debris were described in the response to Question 3.C.1 above.

(3) **What percentage of the sump screen will be blocked by debris or by debris in combination with other material? What is the basis for this answer?**

NMC Response:

As was discussed in the NMC responses to Questions 3.A and 3.B, analytical treatment consistent with the current PBNP licensing bases for debris, other than containment debris, found none that are transportable to the containment sump screens. Therefore, an aggregate effect is not indicated. In accordance with the analysis, since none of the containment debris is transportable to the sump screens, blockage of the sump screens due to miscellaneous containment debris is not anticipated.

OPR 162 demonstrates an additional margin of safety by assuming, consistent with the original design of the screens, that 50% of the submerged area is not available due to blockage. Although we expect no blockage, in OPR 162, consistent with the original licensing basis, we conservatively assume 50% screen blockage.

D. Sump Blockage:

(1) **What are the safety functions of the emergency core cooling system (ECCS) sump? What is the basis for this answer?**

NMC Response:

The ECCS sump (a) serves as the suction source for the RHR pumps during the recirculation phase of a LOCA; and (b) precludes the passage of particulate debris greater than 1/8” in diameter to downstream components, such as the RHR pumps and reactor core.

The first function ensures a continued source of water for core cooling during the immediate and long-term post-LOCA recirculation phase. In fulfilling this first function, the sump serves as a collection point for spilled coolant, injected water, and containment spray run-off. The sump also ensures that excessive air entrainment does not occur, and that frictional head losses through the sump structure are low enough that adequate net positive suction head (NPSH) to the RHR pumps is assured.

The second function is to ensure that the functioning of critical downstream components is not jeopardized by debris suspended in the recirculation flow stream. The establishment of the 1/8” size was originally predicated on the 3/8” diameter containment spray system nozzles. While the spray system is not required to function during recirculation under the current license bases, retention of the maximum particulate debris size is appropriate to ensure operation of other downstream components (the NMC response to Question 3.E provides further discussion of this aspect).

(2) What percentage of the sump screen will be blocked by coatings, insulation, and debris? What is the basis for this answer?

NMC Response:

Blockage from such debris is not anticipated. As discussed in the responses to Questions 3.A, 3.B, and 3.C above, the characteristics of the debris type postulated and the very low fluid velocities preclude the transport of such debris to the sump screens. The only potential debris source that could pose a challenge to the screens are acceptable coatings that have degraded by delaminating or de-bonding. If present within a very limited ZOI/ZOT immediately surrounding the sump screens, the chips or flakes that such coatings could shed would be available to embed on the screen surface. The most recently completed coatings program inspections, together with Engineering Evaluation 2005-0024, show such degraded coatings currently do not exist in the area of interest. Without a viable transport mechanism, the sump screens would remain unblocked by the postulated debris. Note that OPR 162 uses a license basis 50% screen blockage.

(3) What percentage of the sump screen is required to be unblocked (or, what head loss can be sustained) to fulfill its safety functions? What is the basis for this answer?

NMC Response:

The head loss that can be sustained is 1.6’ (19”). This is limited by the potential for direct air ingestion due to the partially submerged screens. The minimum screen submergence at switchover to recirculation is 3.22’. The average submergence is 1.6’.

As the containment continues to fill following a DBA, this minimum sustainable head loss likewise increases. With an expected final containment sump level of 60”, the average screen submergence increases to 30”.

The above direct result is complicated by the recognition that a relatively small “debris collar” around the base of the sump screens could cause a significant reduction in the minimum flow path area. As discussed in the NMC responses to Questions 3.A, B and C, no debris is expected to be transported to the screen and there is no mechanistic basis for positing the formation of such a collar. However, if a debris collar is assumed to form, the collar could cause a significant increase in frictional head loss and creates the potential for flashing if saturated fluid is assumed.

Due to the configuration of the existing sump screens and their close proximity to the sump outlet valve discs, a relatively small accumulation of debris at the base of the screens could cause a disproportionate amount of head loss in the ECCS suction piping.

This condition was evaluated in OPR 162, Revision 1. The findings of OPR 161, Revision 1 are summarized below. For a depiction of the flow details and the calculations involved, please refer to OPR 161, Revision 1, contained on the provided CD-ROM.

OPR 161, Revision 1, determined that, with a “debris collar” around the bottom ~2.5” or more of the sump screen, all flow would be diverted through a small (~3/4” wide) annulus with ~12” inner diameter. The effect of the “debris collar” was to increase the hydraulic frictional losses by an additional 4.8’ under the maximum permissible flow rate (1582 gpm). Existing calculations had previously determined that the NPSH margin available at the same flow rate would be 10.64’, and neglected the 3.22’ of submergence. The net effect is that there is 9’of NPSH margin in excess of the RHR pump requirements, even with the lower few inches already 100% occluded by postulated debris.

Therefore, while the potential detrimental effect of a “debris collar” has been recognized, the effect does not result in a reduction of

ECCS capability beyond that already inherent by the partially submerged screen configuration. As indicated by the responses to Questions 3.A through 3.C above, no blockage of the sump screens by debris is expected. The results of OPR 162, Revision 2, indicate additional margin to accommodate debris, even when none is expected.

(4) **Is there a reasonable expectation that the sump will fulfill its safety function? What are the major uncertainties and the sensitivity of the answer to those uncertainties? What is the basis for this answer?**

NMC Response:

There is reasonable expectation that the sump will fulfill its safety functions. The responses to Questions 3.A, 3.B, 3.C, and Questions 3.D.(1) through (3) show that there is a high degree of confidence that the postulated debris types do not pose a challenge to the ability of the ECCS sump to perform its safety functions. This is based on regulatory guidance supporting the inability of low velocity fluid fields to transport negatively buoyant objects.

Additionally, the quiescent period between the blowdown transient and the initiation of sump recirculation provides time for initially suspended debris to settle to the floor of the large pool, whereupon it would not be available for subsequent transport to the sump screens.

Uncertainties in the quantities and specific mix of debris that could be generated by various sizes and locations of LOCAs are not significant because the debris would be negatively buoyant to the degree that they would behave similarly for any break location.

Since the highest expected flow velocities are less than 0.1 fps, and the minimum velocity necessary to transport debris with a specific gravity of 1.05 or greater is twice this value defined in RG 1.82 Revisions 0 and 1, and NUREG/CR-6773, Appendix B, a margin of at least two (2) exists to accommodate uncertainties in calculated flow velocities.

Additionally, as discussed in the response to Question 3.C.(1) above, the lightest potential debris source is a tape with a specific gravity of 1.1. This is a 100% increase in negative buoyancy beyond the threshold specific gravity of 1.05 cited in RG 1.82, Revisions 0 and 1. The other debris sources considered have substantially higher specific gravities. This represents another conservative factor of two (2) that can accommodate uncertainties in the measurement of this minor debris constituent, and a much larger conservatism for the other debris types considered.

In aggregate, the conclusions of non-transportability and screen operability are based on a foundation of empirical evidence and established regulatory guidance. Significant uncertainties are limited to the exact flow field velocities. However, the hydraulics surrounding the sump screens are not complicated by convoluted flow passages, and the limiting flow rates are based on pump capacities and system hydraulic resistances. As a result, it is estimated that the uncertainties in flow rate (and therefore velocity) are on the order of 10%. With the margins described above, these considerations are enveloped.

E. Affects on Downstream Components:

(1) What types, particle sizes and quantity of materials are expected to pass through the sump screens? What is the basis for this answer?

<u>NMC Response</u>:

Prior to resolution of concerns related to GSI-191, the types of debris explicitly evaluated to pass through the sump screens have been limited to fragments of disintegrated coatings. These evaluations are contained in Section 9 of the unit-specific 1989-90 consultant reports, and in Engineering Evaluation 2005-0024, Revision 1. The various evaluations estimated the total quantity of debris fines that pass through the screens and reach the reactor vessel to be from less than 10 ft^3 to up to 27.5 ft^3. These particles have been estimated to have sizes ranging from 10 microns to 0.125” (the size of the ECCS screen perforations). .

<u>1989-90 Evaluations</u>: These evaluations cite the following assumptions when considering the potential for transport of failed coating fines:

- All coatings inside containment fail (353,100 ft^2).
- The failed coatings have a particle size distribution ranging from 10 microns to 1.0”, with the peak at 0.5”.
- The transport velocities of the fines can be calculated using the same methods as that used for larger particles.

In assessing the concentration of these fines within the recirculating fluid, it was estimated that they would be less than 0.1%. When evaluating the potential for the accumulation of these fines in the reactor vessel, two additional assumptions are made:

- Coating particles less than 0.015” in size reach the sump screens from far-field transport.
- Coating particles less than 1/8” which reach the near sump screen zone (ZOI/ZOT) are available for transport to the ECCS and reactor vessel.

These evaluations estimated that the quantity of debris fines which can pass through the sump screens and reach the reactor vessel would be less than 10 ft^3.

Engineering Evaluation 2005-0024, Revision 1: This evaluation assumed disintegration of 100% of all unqualified coatings inside containment to fines with a size range of 1128 microns or less. The total quantity of unqualified coatings assumed was a bounding figure of 22,000 ft^2. It was further assumed that because of the small size of the particles, 100% remained in suspension and passed through the sump screens.

In performing this evaluation, it was further acknowledged that some of the unqualified (and presumed alkyd) coatings may be epoxy coatings that would not be susceptible to disintegration. However, assuming failure of all the coatings not known to be acceptable would result in a conservatively high calculated concentration of suspended fines. The total volume of the fines was determined to be 27.5 ft^3, giving a volumetric fraction of ~0.13%.

Comparison of Evaluations: Both the 1989-90 reports and Engineering Evaluation 2005-0024 found comparable quantities of suspended fines (same order of magnitude; differing by a factor of 2.8, despite differing approaches to the question). The sizes of the particles were also comparable; all particles <0.015” in the 1989-90 reports, and <1128 microns (0.044”) in Engineering Evaluation 2005-0024.

(2) What ECCS equipment/components have tight clearances that could potentially be affected by foreign materials that pass through the sump screens (e.g., pump seals, flow orifices, throttle valve trim, etc.)? What is the basis for this answer?

NMC Response:

There are no ECCS components that have tight clearances and/or materials that could be unacceptably degraded by foreign materials that pass through the sump screens. This conclusion is based on evaluations performed by the consultant in 1989-90 and Engineering Evaluation 2005-0024, Revision 1.

1989-90 Consultant Evaluations: These evaluations explicitly considered four different aspects of potential effects of suspended debris in the recirculating ECCS fluid:

- Blockage of fluid systems (Section 9.2)
- Effect of Abrasives in the Coatings Debris (Section 9.3)
- Debris Accumulation in the Reactor Vessel (Section 9.5)
- Potential for Core Blockage (Section 9.6)

Components specifically addressed were the containment spray system nozzles, the RHR pumps, SI pumps, containment spray pumps, the reactor vessel, and the fuel assemblies. In each case, the conclusions were favorable, in that:

- The spray nozzles are considerably larger than the maximum debris size postulated.
- The pumps have hard-wear bearing surfaces that will exhibit low wear rates.
- The concentration of coatings debris was estimated to be below the threshold of 0.1% established in NUREG/CR-2792 for negligible effect on pump performance.
- The reactor vessel has considerable free volume in the vessel lower plenum to accommodate accumulated debris (>300 ft^3).
- At 0.15”, the passages in the grid plates of the fuel assemblies have dimensions greater than the 0.125” screen perforation size.

Engineering Evaluation 2005-0024: This evaluation relied on information compiled for evaluation of components for downstream effects under the on-going effort to resolve concerns related to GSI-191. It determined that failure of the mechanical seals on the RHR and SI pumps from operation with suspended coating decomposition particles is not expected. The same design seals are used in applications with similar debris laden fluid such as pulp and paper, petrochemical, food processing, and waste water treatment.

The evaluation concluded that the orifices (flow instrumentation and flow limiting orifices) in the credited ECCS flow paths are on the order of inches, and therefore, are not subject to blockage by the fine particulates. Also, after sump recirculation is established, valves in the flow path are not relied upon to reposition. The valves in these flow paths are 2” or greater in size, and have stainless steel or harder wearing surfaces. While it may be desirable to throttle the 8” diameter RHR heat exchanger outlet butterfly valves, these large diameter valves are not expected to be susceptible to significant degradation from suspended particulates. Based on these considerations; wear, erosion and blockage of valve components are not a factor.

The evaluation also concluded, as did the 1989-90 reports, that the reactor vessel and core flow passages are on the order of fractional inches or more, and are not susceptible to fouling by the fine particulates.

4. SI-850 Valves

A. What are the safety functions of the valves (e.g., to open/stay open, to shut/stay shut, to maintain leak tightness) and what is the basis for this determination? What are the ECCS pump minimum and maximum recirculation flows and net positive suction head (NPSH) requirements? What is the basis for this answer?

NMC Response:

The SI-850A&B have a safety function in both the open and shut directions. Safety-related systems, structures and components (SSCs) include SSCs that are relied upon to remain functional during and following design basis events to assure the capability to shut down the reactor and maintain it in a safe shutdown condition.

The ECCS system is designed such that the failure of a single active component or the failure of a passive component during the long-term cooling period does not interfere with the ability to meet the necessary long-term cooling objectives. The RHR system is designed to provide the following safety-related functions:

- Deliver borated cooling water to the RCS during the injection phase of SI
- Recirculate and cool the water that is collected in the containment sump and return it to the RCS during the recirculation phase of SI
- Provide the means to preclude containment leakage through the RHR system piping penetrations following accidents

- For piping and components that are part of the reactor coolant pressure boundary, maintain pressure boundary integrity during all modes of plant operation

Recirculating and cooling water that is collected in the containment sump and returning it to the RCS requires the SI-850 valves to have a safety function to open and stay open. Since the SI-850 valves are the only installed valves that can isolate a passive failure of the piping between containment and the 1(2)SI-851A&B, the SI-850 valves have a safety function to shut and remain shut to minimize the effect of a passive component failure through the RHR system post-accident. This leakage is a passive failure of one suction line (excessive packing or weld leakage) and will not impair the operation of the redundant valve. Shutting the SI-850 valve for the affected train stops excessive leakage during the long-term cooling, and therefore, this excessive leakage cannot interfere with the other system from performing its long-term cooling objectives.

Dose consequences and the licensing basis are addressed in Question 5.E.

The industry definition of passive failure evolved during PBNP's original licensing and culminated in the redesign of the ECCS system, including the inclusion of the SI-850 valves. The SI-850 valves were installed specifically to isolate the ECCS line to "minimize" the effect of a passive component failure.

<u>Containment Sump Outlet Flows (Recirculation Flows)</u>: At the initiation of containment sump recirculation, a single operating RHR pump's suction is switched from the RWST to the containment sump. The flow rate from the sump would be bounded by a maximum of 1582 gpm. The operating containment spray pump(s) would continue to draw down the RWST inventory until the criteria to secure the pumps is reached. The SI pumps would be secured once it was verified that the RHR pumps were providing adequate injection flow (this action prolongs the period that the RWST is available for injection).

The flows that could be provided under other alignments have been analyzed, but these flows are not procedurally permitted because adequate NPSH is not available to the RHR pumps. RHR flow is injected into the core through the core deluge nozzles. Therefore, as long as the injection flow is greater than the core cooling flow requirement, the core will receive adequate cooling and the excess will be diverted out the RCS break. The procedurally limited flow of 1560 gpm exceeds the core cooling required flow at the start of recirculation. The procedurally limited flow is a result of NPSH concerns as described in OPR 162. Resolution of this OPR will address this condition.

<u>Boron Precipitation Control</u>: The flow values contained in the emergency operating procedures (EOPs) are maximum allowable flows and not minimums. Boron precipitation control is obtained at a reduced maximum flow of 1560 gpm. In the May 7, 1975, licensee response to an NRC letter

dated March 14, 1975, on this subject, ECCS long-term cooling requirements were provided. The evaluation was based on the generic Westinghouse evaluation, "Long Term Cooling – Boron Considerations." This submittal stated that flow from a single pump (187 lbs/sec or ~1350 gpm) during a large break LOCA was more than adequate to prevent boron precipitation.

Net Positive Suction Head: Recirculation operation gives the limiting NPSH requirement. The available NPSH is determined from the containment water level, and the pressure drop in the suction piping from the sump to the pumps. The RHR pumps need at least ~8' of NPSH at 1582 gpm. The RHR pumps are located at El. -19'3" to assure the necessary NPSH at the pump suction when the recirculating water is at 212°F with atmospheric pressure in the containment.

B. Have the valves been adequately tested to demonstrate that they will perform each of their safety functions identified above? Explain and identify what testing has been performed? What is the frequency of this testing and how do the test acceptance criteria demonstrate/relate to the valve safety function? What is the basis for this answer?

NMC Response:

The SI-850 valves have always had a safety function to open and are tested to ensure the open function is maintained as discussed below. During the review of the safety functions for the valves (see response to Question 4.A), it was identified that the valves also have a safety function to shut. A corrective action program document was initiated since the valves were not currently credited in the PBNP inservice testing program as performing a safety function in the shut direction. The shut safety function testing has been incorporated into IST program. Seat leakage testing that would be required for a safety-related function in the shut direction is addressed in the response to Question 5.I.

Open Safety Function: These normally shut, hydraulically-operated valves are located inside containment in the line leading from Sump "B" to the suction of RHR pumps. The valves perform an active safety function in the open position. The SI-850 valves must be capable of opening, by remote manual switch actuation, when transitioning from the injection mode of SI to the recirculation mode of SI. When the initial supply source of SI water from the RWST is effectively depleted following a LOCA, suction for the SI and RHR pumps is switched from the RWST to the containment sump to provide long-term core cooling. The SI-850 valves receive no automatic actuation signals to open and must be aligned from the control room using their associated control switches. They have no maximum design stroke time limits associated with the safety function in the open position and fail "as-is" on a loss of power.

The SI-850 valves are tested quarterly in the open direction via the inservice testing (IST) program using Inservice Test (IT) 40, "Safety Injection Valves (Quarterly) Unit 1", and IT 45 "Safety Injection Valves (Quarterly) Unit 2".

ASME OM Code Paragraph ISTC 4.2.8 provides the basis for IST acceptance criteria for open stroke time and for position indication verification (PIV) testing.

Valve stroke time acceptance criteria are based on ASME OM Code, which sets the acceptance criteria at the baseline reference value +/- 50%. IT 40 and IT 45 check the output pressure of the SI-850A(B) hydraulic pumps in both the open and shut valve stroke directions. PBNP currently requires that the hydraulic pressure not exceed 1500 psig when shutting the valve to prevent damage to the hydraulic operator. In the open direction, the hydraulic pressure for each valve is required to be between 1150 and 1500 psig. The 1150 psig lower limit is to ensure that the hydraulic operator is capable of opening the valve against the weight of the valve, packing friction, head of the containment sump and post-accident containment pressure.

Shut Safety Function: The valves had previously not been credited with a having a shut safety function. This is documented in a corrective action program document. The valves' shut stroke time and shut position indication verification is performed under IT 40 and IT 45 for trending of valve degradation using the same guidance of ASME OM Code Paragraph ISTC 4.2.8 as the open direction test.

The response to Question 5.I documents the allowable leakage requirements of the SI-850s in the shut position.

C. Is there a reasonable expectation that the valves will perform their safety functions for the duration of the events, as defined in the safety analyses? What is the basis for this answer?

NMC Response:

There is a reasonable expectation that the SI-850A(B) valves (containment sump "B" isolation) will be able to perform their safety functions for the duration of the events as described in the safety analyses.

The design basis for these valves has previously identified that the valves only had an open active safety function. CAP 069891 has identified that an active safety function in the shut direction also applied.

The ECCS system is designed such that the failure of any single active component or the failure of a passive component during the long-term cooling period does not interfere with the ability to meet the necessary long-term cooling objectives.

The PBNP licensing basis assumes a passive failure or an active failure during the long-term cooling phase. Either an active or a passive failure would remove one train of ECCS, leaving the other train operable to recirculate and cool the water that is collected in the containment sump and return it to the RCS. Therefore, the only credible mechanisms to prevent the SI-850 valves from performing their safety function would be environmental qualification considerations.

The containment sump isolation valves have hydraulic cylinders for opening and shutting the valves, which are mounted directly to the piping in the containment tendon gallery. The hydraulic pumps which provide pressure to the cylinders for operation are mounted in the PAB pipeway hallway. A review of the valves' ability to perform intended design functions was performed. The review determined that the hydraulic cylinder components pressurized to shut the valve and the hydraulic units located in the PAB pipeway meet the required service conditions for valve operation. It was determined that the containment sump "B" isolation valves are not in compliance with the environmental qualification (EQ) program requirements. As evaluated in OPR 171, SI-850A(B) are capable of performing their safety-related function throughout the recirculation phase.

Presuming one SI-850 valve was shut to minimize the effect of a passive component failure, the valve would remain in the shut position. The forces from the containment recirculation sump liquid level in addition to a containment overpressure would maintain the valve shut. The remaining valve is maintained opened to ensure one train of RHR is in operation for core cooling.

The open containment sump isolation valve needs to remain open to provide a suction path for the RHR pump. The valve and its operator are designed for the post-accident environmental conditions in the PAB pipeway hallway and the tendon gallery. The system was designed to mitigate either an active or a passive failure during recirculation operation.

In the event of a single active or passive failure during long term cooling the second train of RHR would be placed in service to ensure core cooling is maintained.

The potential for valve drift and resulting effects are discussed in the response to Question 4.D.

D. What are the consequences if the valves fail closed? If the valves fail closed, can they be re-opened? If these valves can drift shut, what amount of closure will cause the open indication in the control room to be lost and what will be the effect on recirculation flow/NPSH/pump operation with these valves in this partially closed position? Is the equipment that provides control room position indication qualified in accordance with 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants? If they can be re-opened what are the consequences of the time period the valves are not fully open? What is the justification for the time period assumed? Can the valves be opened with pumps in operation? What is the basis for this answer?

NMC Response:

Consequences of Valve Closure: Failure of both valves to open would require two active failures, which is outside the design basis for PBNP. Failure of one valve to open is considered the single active failure of the RHR system and the second train of the RHR system would be started to provide containment sump recirculation. The valves in one train are verified as open prior to establishing containment sump recirculation using plant procedures. There are no common cause failures that could affect both trains.

The hydraulic units located in the PAB pipeway are environmentally qualified and would allow remote operation of the valves from the control room during the complete duration of a LOCA event requiring long-term recirculation.

Failure of either containment sump "B" isolation valve to open requires operator action to manually open the valves using a staged hydraulic hand pump. If containment sump recirculation can not be established, there is procedural guidance which directs operators to utilize contingency actions. Additional information regarding radiological considerations for local operation is provided in the response to Question 4.E.

The SI-850A/B valves can be reopened if they move from the open position to an intermediate or shut position. The valves can be remotely opened, or the staged hand hydraulic pump can be used in accordance with established emergency operating procedures and contingency actions.

Use of the hydraulic operator requires placing the associated valve's control room hand switch to open position to reestablish the hydraulic force to open the valve. This is an expected response of the control operator to the valve being out of position as specified per procedures.

Control Room Indication: The valve travel distance that will bring in the control room intermediate indication from the open valve position was measured via performance of work orders that obtained stroke distances. The calculation performed for OPR 162 shows that there will be adequate NPSH for the RHR pumps when an SI-850A (B) valve is at an intermediate indicated valve position. OPR 162 also established that there would be adequate time to take remedial action, such as reopening the valve, between indications of valve drifting and the loss of the RHR pump suction source.

Environmental Qualification: The equipment necessary for the control room to remotely observe valve position and operate the SI-850A(B) valves during the duration of the recirculation phase is capable of operating within the environment expected. EQ documentation is currently nonconforming and will be upgraded. See the response to Question 1.C for additional detail.

Consequences of Partial Valve Closure: There is not a defined drift closure time period or inadvertent closure period assumed for the valves with RHR in operation either in the licensing or the design basis for PBNP. Failure of one valve to open is considered the single active failure and the second train of containment sump recirculation would be started. The valves are verified as open prior to establishing containment sump recirculation.

A corrective action program document was initiated to investigate the potential for a recirculation valve to drift shut. Closure of the valve would impact the NPSH to the recirculation RHR pumps as the friction factor of the water flowing to the RHR pump suction would increase as the valve drifted shut. Drifting of the valve to the shut position would be apparent to operating personnel as a loss of RHR pump flow via main control board indication. Sump “B” recirculation would be restored in accordance with approved plant procedures.

Each containment sump “B” isolation valve has position indication in the control room with red and green position indicating lights adjacent to the control switches. Changes in the status of this indication would be apparent to operators. Prior to assuming control room duties each shift, and frequently during the shift, licensed operators are required to perform a main control board walkdown that would identify potential changes in valve position indication.

A corrective action program document was initiated to investigate potential hydraulic fluid leakage that could affect SI-850A(B) valve drift. Hydraulic fluid leakage paths to consider would be past the hydraulic cylinder internal piston ring, out of the system, or through the solenoid-operated pilot valve under the influence of gravity and the differential pressure generated across the valve by flowing sump or RHR fluid. A qualitative assessment of leakage is provided below based on the valve operator design.

The impact of internal piston ring leakage would be expected to be a long-term degradation mechanism as minor leakage from the hydraulic cylinder could be postulated through the hydraulic seals to the environment. Gross leakage is not assumed as it would be observed during functional testing since the stroke time of the valve would change as a result of this failure mechanism. Failure of the valve hydraulics would be considered an "active" failure and would apply to only one of the two valves.

The hydraulic system solenoid-operated valve utilizes close tolerance metal seating surfaces. Leakage past the hydraulic system solenoid-operated pilot valve would be related to hydraulic pressure created as a result of forces placed on the hydraulic cylinder. The force on the hydraulic cylinder would be generated by the influence of gravity and the differential pressure generated across the SI-850A(B) valves by flowing sump/RHR fluid creating a valve stem force. The SI-850A(B) valve stem load trying to shut the valve (weight of valve, stem ejection forces) verses stem forces maintaining the valve open(piston friction, stem friction) is very small. Therefore, if drift occurs, it would be relatively slow. A drift rate is not definable in the amount of time it would take to cause the intermediate valve indication to actuate.

PBNP has not tested potential drifting of the SI-850A(B) in the closing direction to date. A corrective action program document has been initiated to determine testing methods and to establish acceptance criteria.

In OPR 162, a sensitivity evaluation of NPSH verses valve position was performed. The results indicate that valve drift would have to occur before the partially shut valve would begin to create more of a pressure drop than when it was full open. The maximum possible open stroke is 2.5". Field measurements determined that the valves are set to provide a full open stroke of at least 2". The point at which the valve begins to increase head losses above the acceptable head loss is .85" of open travel. As noted previously, intermediate position indication lights for these valves on the main control boards would show drift by at least 1.25" of open valve travel. Therefore, based upon the control room light indications and industry experience that a hydraulic leak is expected to be slow, licensed operators would recognize potential valve drifting shut prior to impacting core cooling.

<u>Can the Valves be Open with the RHR Pumps in Operation</u>: The containment sump "B" recirculation valves can be opened with the RHR pumps operating when still aligned to the RWST as a suction source. The differential pressure forces on the valves at this time are less than the forces assumed in Calculation 2001-0001, "Hydraulic Pressures Associated with the SI-850 Valves," since RWST water level head will reduce the forces assumed in the calculation.

In the event of valve drift, the SI-850 valves would be capable of stroking from a partially shut position to full open. The forces required to reopen the valve to the full open position are significantly less than maximum loads used in Calculation 2001-0001 since containment pressures would be lower and the differential pressure across the valve disc would be minimal because the valve is still open.

E. What is the radiation exposure to the operator if local manual action is necessary? What is the basis for this answer?

NMC Response:

The SI-850 motor-hydraulic units, referred to as the valve operators, that would be accessed to manually change the position of the SI-850 are located outside Pipeways 2 and 3 in the access gallery for Unit 1 and Unit 2, respectively. The valve operator is shown in the FSAR Figure 6.2-2 and as Attachment 5 of this enclosure. The valve operators for Unit 1 and Unit 2 are on the El. 8' located near 1(2) RK-51/52 Pipeway 2 (3) instrument panels, which are shown on FSAR Figure 1.2-4. During the design and construction of the plant the motor-hydraulic units for each of the SI-850 valves were intentionally located in the PAB such that access post-accident could be made if needed. The SI-850 valves can be operated remotely from the control room.

Since the passive failure of one suction line (presumably excessive packing or weld leakage) will not impair the operation of the redundant valve, multiple failures would have to occur to require local operator action. For example, assuming a failure occurred on the inservice recirculation train, local operator action would be necessary if the operator is unable to isolate the failed train from the control room or the operator is unable to place into service the opposite train. Multiple failures are not taken in conjunction with a design basis event. Therefore, consistent with the design basis for PBNP, access to these valves was not considered required, but was possible, based on the intentional selected physical location.

Dose considerations for local manual action in the PAB post-accident are described in FSAR 11.6 under auxiliary shielding. The auxiliary shielding is based on a design basis LOCA with minimum safeguards that results in a gap release of all of the fuel rods, as determined by the 10 CFR 50.46 evaluation presented in FSAR 14.3.2 and discussed in FSAR 14.3.5. Specifically, FSAR 11.6 states the following:

"All components necessary for the operation of the external recirculation loop following a loss-of-coolant accident are capable of remote manual operation from the control room and can be powered by the emergency diesel-generators so that it should not be necessary to enter the auxiliary building in the vicinity of the recirculation loops."

This section of the FSAR goes on to state that if access is essential to the continued operation of the engineered safeguards system during the recirculation phase dose reduction measures would be applied. Such dose reduction measures would be additional shielding, limited duration and respiratory protection. Estimated dose rates in the vicinity of the RHR recirculation piping are stated as 25 R/hr one-hour, post-accident whereas dose rates on the recirculation loop are stated as 200-300 rem/hr immediately following the initiation of recirculation. The basis for these dose rates is provided in Table 11.6-6 of the FSAR.

In response to NUREG-0578 Item 2.1.6.b, "Design Review of Plant Shielding of Spaces for Post-Accident Operations," reissued as NUREG-0737 Item II.B.2, PBNP re-evaluated the shielding design of the PAB to ensure areas requiring post-accident access were habitable. This review was performed under the assumption of a fuel-melt accident and resulted in several shielding modifications. The location of the containment sump suction isolation valve operators were not identified as vital areas, that is, areas requiring access post-accident; however, these areas were shown to be inaccessible. Access to this area is limited due to the direct radiation from unshielded low pressure safety injection lines transporting liquid from the RHR heat exchanger to the safety injection/containment spray pump room. Acceptance of the implementation of NUREG-0578 Item 2.1.6.b was provided to PBNP on April 9, 1980, when the NRC acknowledged that shielding was generally adequate and additional shielding of the C-59 control panel was under consideration. Permanent and portable shielding was later placed in the area of the C-59 panel as well as other areas of the PAB. This work was communicated to the NRC via responses to NUREG-0737 Item II.B.2. The NRC accepted the NUREG-0737 Item II.B.2 vital access response on November 3, 1983.

Therefore, based on a review of the current licensing basis and design basis of PBNP, local operator action is not necessary to open/shut the containment sump suction valves post-LOCA. This is because they are remote-operated valves and a single failure on one recirculation train will not prevent the other train from performing its design function. Under the presumption of a radiological design basis LOCA (i.e., fuel melt), the location of the valve operators is not accessible due to the unshielded recirculation lines in the vicinity of the operators. However, under the presumption of a design basis LOCA that credits minimal safeguards on injection (i.e., gap release); these areas would be accessible on a limited basis if additional protective measures were taken into consideration.

F. Will flashing occur in the piping below the valves when they are opened to perform their safety function during an event, including the long term? Consider containment overpressure, ECCS flow, and the number of ECCS trains in operation. If containment overpressure is needed, has it been analytically shown that the minimum overpressure assumed in the analysis will be present for the limiting combination of conditions (e.g., including inadvertent operation of secured equipment that could reduce containment pressure), including the long term? What is the basis for this answer?

NMC Response:

Flashing will not occur in the piping below the valves as described in OPR 162 (on the provided CD-ROM). During the recirculation phase containment equilibrium pressure due to the partial pressure from air existing in containment before an accident and a partial pressure from steam at 212°F due to a pool of water at the bottom of containment are credited. However, as discussed in OPR 162, crediting the containment equilibrium pressure is not in conformance with the current licensing basis. To change the conclusions of the OPR, air would need to be removed from containment. The containment structure is designed for the pressure and temperature resulting from a design basis accident; however, a breach of containment is not within the design basis of PBNP.

In order for a large amount of air to be quickly removed from containment, a relatively large opening that can vent air must be made in containment. Large openings that could communicate directly with the atmosphere can be made by inadvertent operation of the purge system, opening of both containment doors, or opening of the fuel transfer canal. The purge system contains blind flanges on both penetrations during Modes 1 to 4 that would need to be removed in order to use the purge penetrations. The containment doors are mechanically interlocked such that only one door at a time can be opened. The fuel transfer canal contains a blind flange that must be removed prior to the use of the penetration. The next largest penetrations are the main steam and main feed penetrations. These penetrations are connected to a closed system inside containment. Therefore it is not likely that large amounts of air could be released by inadvertent operation of secured equipment.

OPR 162 used the most limiting train of RHR for the determination of flashing. Each train has an independent fine screen suction strainer. Cross-connection of both trains of RHR is not an alignment directed by the EOPs. Therefore, both trains cannot draw suction off the same fine strainer, so the flow rate assumed in the OPR is bounding.

The PBNP licensing basis assumes the failure of a single active component or the failure of a passive component during the long-term cooling period. Inadvertent operation of a second train is not within the design basis for this system. However as stated above, if the second train was started, it would draw from its own strainer and SI-850 valve, and would not have an effect on the flashing considerations.

G. If flashing occurs, what are the potential consequences? What is the basis for this answer?

<u>NMC Response</u>:

Flashing does not occur as documented in OPR-162 and discussed in the previous question.

5. ECCS Leakage from the Recirculation Line (flange/body-bonnet/packing/weld)

A. What is a technically defensible failure (leakage rate) to consider and when and where are these leaks postulated to occur? What is the basis for this answer?

<u>NMC Response</u>:

As defined in FSAR 6.2, the passive failure of one suction line is assumed to be due to excessive packing or weld leakage that will not impair the operation of the redundant recirculation train. This FSAR section also indicates that a RHR pump seal failure rate is 50 gpm.

During normal plant operation, the leakage limit from the ECCS is maintained to be 400 cc/min or less. This 400 cc/min value is constrained by the control room dose analyses and is described in FSAR Sections 6.2 and 14.3.5. The control room dose analyses assumed an ECCS leak rate of 400 cc/min, for 30 days following an accident.

The final form of the current radiological analysis for control room habitability was communicated by NMC to the NRC on June 3, 1997. This submittal provided additional information as a basis for the exclusion of a passive failure post-LOCA. The analyses of record were approved in a Safety Evaluation Report dated July 9, 1997, "Issuance of Amendments Re: Technical Specifications Changes for Revised System Requirements to Ensure Post-Accident Containment Cooling Capability." The primary basis for the exclusion of passive failure was the assertion that radiological dose post-LOCA for PBNP had not previously assumed a passive failure in conjunction with the design basis radiological analysis. The only assumed failure for the LOCA radiological design basis dose analysis has been the loss of an emergency diesel generator, which limits the containment spray and ventilation systems to one train each.

The credible leak sources for this type of leak consist of a malfunctioning residual heat removal pump seal, flange gasket, or a valve with degraded packing. The flow rate from any one of these sources will be less than 50 gpm. Original Technical Specification 15.4.4 for PBNP (April 1970) stated, “The limiting leakage rates from the residual heat removal system are a judgment value primarily based on assuring that the components could operate without mechanical failure for a period on the order of

200 days after a design basis accident." This value was used in Chapter 14 of the Final Facility Description and Safety Analysis Report (FFDSAR).

During the recirculation phase, continuous ECCS leakage may become airborne and escape through the PAB vent stack to the environment. This leakage is not expected to exceed 400 cc/min. Radiological analyses of offsite dose due to this leakage have conservatively doubled the expected ECCS leak rate; assuming a combined ECCS leak rate of 800 cc/min during the accident. Offsite radiological consequences of the LOCA, including this ECCS leakage, are described in FSAR Section 14.3.5. The 50 gpm passive failure is not included in the dose analysis, since this leakage is expected to occur around 200 days following an accident, which is after the 30 days assumed in the offsite and control room dose analysis.

To maintain the leakage limit of 400 cc/min or less for the dose analyses, a series of Leakage Reduction and Preventive Maintenance (LRPM) tests are performed during each refueling outage. These tests measure and quantify the leakage from the system to the atmosphere by looking at leakage from individual components outside containment (i.e., valves, body-to-bonnet joints, packing) and portions of trains or systems. Seat leakage at boundary valves is included in the total leakage value as leakage to other systems may ultimately be exposed to atmosphere. The leakage determined in these tests is collected at conservatively higher test pressures than would be experienced during a design basis event.

Leakage–to-atmosphere in the LRPM program is maintained "as low as reasonably achievable." Active leakage, typically on the order of drops per minute, is corrected prior to completion of a refueling outage. Acceptance of an active leak requires a corrective action program document to be initiated and the active leak evaluated as acceptable for unit restart.

In summary, the design basis leak rate for a passive failure in the ECCS containment suction line is 50 gpm. This leak is the expected worst case for a RHR pump seal failure that bounds all other leakage in the suction line through packing or weld leakage. This passive failure is not included in radiological analyses as currently defined in PBNP licensing basis. The time for such a passive failure to occur is on the order of 200 days following a design basis accident.

B. What compensatory measures are available to detect and isolate this leakage? If non-safety related equipment is relied on to support detection and isolation explain why this is appropriate. What is the basis for this answer?

<u>NMC Response</u>:

There are three general areas where a passive failure in the containment sump recirculation line to the RHR pumps could occur: The tendon gallery; the RHR system pipeways in the PAB; and the RHR pump compartments.

<u>Leakage resulting from a passive failure in the tendon gallery</u>: The plant design is such that leakage in the tendon gallery would have a flow path to the "A" RHR pump compartment and would be detected by the level transmitter in this compartment. However, it was discovered that the tendon gallery sleeves are grouted closed and this leak path is currently not available. CAP 069723, "Design Basis Leakage Detection Capability May Have been Defeated," was submitted on January 10, 2006, in response to this discovery. OPR 170 concluded this condition as operable but nonconforming. The OPR determined that sufficient time and containment sump volume is available for detection by means of control board indications prior to challenging core cooling post-accident. .

If the leakage from the ECCS system to within the tendon gallery occurred, the means for detection of this leakage would depend in part on which systems are operable/functional post-event. A nonsafety-related sump pump automatically starts when tendon gallery water level increases. The tendon gallery sump pump automatically pumps water to the façade sump. As long as the tendon gallery sump pump is functional, plant operators would receive a façade sump alarm. When this alarm is received, the sump is pumped out using approved plant procedures, 1(2)-SOP-WL-002, "Pumping Façade Sump Unit 1(2)." Sump samples are taken prior to pumping the façade sumps. Adverse chemistry results would prompt an immediate investigation by operators, as skill of the craft, into the source of the leak so the leak location could be identified. Similarly, if repeated pumping of the sump occurred over a short period of time, an immediate investigation would be initiated into the source of the leak.

A passive leak would result in a reduction of containment sump "B" level. During recirculation, the two safety-related redundant containment sump B level transmitters are monitored, so the control room staff would detect a gross change in the containment sump "B" level. A gross change in the containment sump "B" level would be noticed within at least one shift. Once a gross change in the containment sump "B" level has been observed, an immediate investigation into the source of the level change would be initiated by the control room staff.

OPR 170, Design Basis Leakage Detection Capability May Have Been Defeated, and OPR 171, Safety Function for Containment Sump "B" Isolation Valves, demonstrate that despite a passive failure in the recirculation lines of the RHR system, the safety function of the system can be maintained. Isolation of the passive failure would be accomplished by shutting the SI-850 and SI-851 valves on the failed train of containment sump recirculation.

If the tendon gallery sump pumps failed or would not function, the tendon gallery could potentially flood up to the façade floor (El. 6'-6") before the leak was detected and isolated. Filling the tendon gallery would take about 82,400 gallons of water. If the bounding leak rate of 50 gpm was located somewhere within the tendon gallery, it would take approximately 27 hours to fill the tendon gallery with water to the façade floor. Additional detail for time availability for response to this failure is provided in the response to Question 5.C.

Leakage resulting from a passive failure in the RHR valve pipeway in the PAB: Leakage into the RHR valve pipeway would reach the RHR pump compartments and would be detected by the non-safety related level transmitters in the RHR pump compartments. Leakage in the RHR pipeways would collect on the floor in the pipeways, which are located behind the RHR pump compartments. It would then drain through the associated RHR pump compartment wall into the RHR pump compartment. This pipeway is divided into two sections by a 7' wall. At the bottom of each of these sections of the pipeway there is a 4" square hole that runs through the RHR pump compartment wall into the RHR pump compartments. Both SI-851 valves are located directly behind the "A" RHR pump compartment in the RHR pipeway and leakage from the "A" and/or "B" train upstream of the SI-851 valves will show up in the "A" RHR pump compartment. Once leakage drains into the RHR pipeway, it would be handled as if it was leakage in a RHR pump compartment.

Leakage resulting from a passive failure in the RHR pump compartments: Leakage detection in the RHR pipeways and RHR pump compartments would be achieved through the use of sump level detection. Leakage that reached the RHR pump compartments, either from the RHR pipeway or from within the RHR pump compartments, would be detected by the nonsafety-related level transmitter installed in each of the RHR pump compartments. As a result, an RHR pump room high-level alarm would be indicated on the control room main control boards. Each RHR pump compartment is equipped with a floor drain and separated equipment drains. The floor drain from each RHR pump compartment flows through an individual pipe to the El. -19' PAB sump. Two 75-gpm sump pumps transfer the leakage collected in this sump to the waste disposal system for processing. The supply and discharge piping and valves for the RHR pumps are located in a pipeway adjacent to the pump compartments.

Procedural guidance for detection and isolation of a leak that flows into the RHR pump compartment is provided in emergency operating procedure (EOP) EOP-1.3, "Transfer to Containment Sump Recirculation Low Head Injection." This procedure first detects which RHR pump compartment is affected by use of the individual level indicators located in each pump compartment. The procedure then directs the operator to open the affected drain valve to the El. -19' PAB sump. Once this drain is opened, the frequency of operation of the -19' PAB sump pumps is monitored to attempt to quantify the rate of the leakage from the passive failure in the recirculation line. If needed, the RHR pump in the train with leakage would be shut down and isolated by closing the SI-850 and SI-851 valves for that train. This prevents gross diversion of containment sump inventory through the failed containment sump recirculation line.

The level transmitters, level switches and sump pumps used to detect a passive failure within the PAB (RHR pipeway or RHR pump compartment) use nonsafety-related power supplies and components. However, a safety-related bus, through a nonsafety-related power panel, powers the RHR pump compartment level switches. This arrangement provides reasonable assurance the level transmitters will be available to detect a flooding concern within the RHR pump area.

The RHR pump compartment drain isolation valves are also powered from nonsafety-related buses, which in turn, are powered by safety-related buses. Again, this arrangement provides reasonable assurance the drain isolation valves will be available to mitigate a flooding concern within the RHR pump area.

In addition, the RHR pump compartment level switches are manually lifted each quarter to assure that they are working properly and producing control room alarm and indication. The RHR pump compartment drain isolation valves are also operated quarterly from the control room to assure that they are functioning properly.

The El. -19' PAB sump pumps are powered from two independent power supplies; one from a Unit 1 power supply and one from a Unit 2 power supply. While they are powered from nonsafety-related buses, these buses are powered off safety-related buses (2B03 and 1B04) that have emergency diesel generator supplies. During an accident, the nonsafety-related buses receive a safety-injection stripping signal. However, as the accident progresses into the recovery phase and safeguards electrical demand decreases, operators would be able to reenergize the stripped bus, as needed, to support flood mitigation concerns within the RHR pump area. For a design basis LOCA coincident with a loss of offsite power, both El. -19' PAB sump pumps would have power stripped. PAB sump level detection, however, would remain energized, thus prompting operators to reenergize the sump pump power supplies, if needed. EOPs direct the motor control center for the PAB sump pumps to be restored and the PAB sump level to be monitored.

As discussed above, isolation of a passive failure in the containment sump recirculation lines would be accomplished by shutting the SI-850A(B) and SI-851A(B) valves. The SI-850 valve discs are located inside the containment. A dedicated hydraulic pump located in the PAB is used to control a hydraulic cylinder located in the tendon gallery, which opens and shuts the valve. The downstream SI-851 valves are motor-operated gate valves located in the PAB. Both the SI-850 and SI-851 valves are considered to perform an active safety-related function in both the open and shut directions.

The NMC response to Question 5.H discusses the recent change in safety classification associated with the SI-850 valves in the shut direction. Both the SI-850 and SI-851 valves are included in the IST program and are tested quarterly in accordance with the ASME OM Code. A review of test data confirmed that none of these valves experienced an inservice testing failure over the last fuel cycle that would challenge the ability of the valve to perform its intended safety functions.

In addition to the inservice testing program, PBNP has several other programs in place to assure that the containment sump recirculation lines and associated components are capable of performing their intended safety functions. The programs include testing the emergency core cooling system (ECCS) via the Leakage Reduction and Preventive Maintenance (LRPM) program. Leakage from the ECCS recirculation line is routinely checked and monitored during the performance of the LRPM tests on a refueling outage frequency. The Units 1 and 2 LRPM databases are maintained and updated during the performance of the LRPM tests with the total ECCS leakage being recorded. The total ECCS system leakage is verified to be less then the FSAR Chapter 6.2 limit of 400 cc/min.

PBNP's preventive maintenance program also supports the reliability of the SI-850 and SI-851 valves. The valve operators for the SI-850 valves are disassembled and inspected every 10.5 years. The operators for the SI-851 valves are diagnostically tested every 4.5 years and are disassembled and inspected every 12 years.

PBNP's design, testing and maintenance programs provide assurance that both the safety-related and nonsafety-related components within the containment sump recirculation lines and the sump systems used to detect and manage leakage, remain capable of performing their functions, including mitigating the consequences of a passive failure within the lines.

C. How long will detection and isolation of a passive leak take? What is the basis for this answer?

NMC Response:

Passive Failure in Tendon Gallery: The tendon gallery sump pump is expected to pump about 5 gpm to the façade sump. The façade sump alarm corresponds to about 482 gallons. Should the façade sump have been emptied immediately prior to the passive failure, an alarm in the control room would be received in less than two hours. It would take longer to identify a leak that is smaller than the tendon gallery sump pump's capacity.

Grouting was discovered between the tendon gallery piping sleeve and pipe. The grouting prevents leakage from the ECCS suction line in the tendon gallery from entering the RHR pump compartment. An operability recommendation (OPR 170) concluded the condition was operable but nonconforming because sufficient time and containment sump volume were available to detect a 50 gpm leak prior to the loss of net positive suction head on the RHR pumps. As a result of this nonconforming condition gross containment sump leakage would be used to identify the leakage source as discussed below.

A passive leak would result in a reduction of containment sump "B" level. During recirculation, operators routinely monitor the two safety-related redundant containment sump "B" level transmitters and would notice a gross change in the containment sump "B" level. A gross change in the containment sump "B" level caused by a 50 gpm leak would be noticed within at least one shift. This is based upon control board reviews and daily log sheets. Once a gross change in the containment sump "B" level has been noticed, an immediate investigation into the source of the level change would be conducted by control room personnel.

Recent operability evaluations (OPR 170 and OPR 171) have demonstrated that despite a passive failure in the recirculation lines of the RHR system, the safety function of the system can be maintained. Isolation of the passive failure would be accomplished by closing the SI-850 and SI-851 valves on the failed train of containment sump recirculation. Containment sump level would approach the minimum NPSH requirements for the RHR pumps in about 57 hours assuming a 50 gpm leak rate. Sufficient time exists between detection (within one shift) and loss of decay heat removal capabilities (about 57 hours) to allow operators to isolate a postulated 50 gpm passive leak.

Passive Failure in RHR Valve Gallery or Pump Cubicle: The original design of the RHR pump compartments and the adjacent compartments are designed so they have a flow path to the RHR pump compartment. These RHR pump compartments are approximately 200 ft^3 in size and will completely fill in about 30 minutes at a flow rate of 50 gpm Additional information on the design of the RHR pump compartments, leakage detection and flow path are contained in the NMC response to Question 5.B.

A passive leak within the RHR pipeway or one of the RHR pump compartments would result in the credible leak source flowing to one of the RHR pump compartments. This would result in an RHR pump compartment high level alarm in the control room. The alarm would require that operators respond as directed by the associated alarm response book (ARB) procedure, which requires that the pump compartment be drained and the leakage source isolated, if possible, to prevent damage to the RHR pumps. Isolation of the passive failure would be accomplished by shutting the SI-850 and SI-851 valves on the failed train of containment sump recirculation.

D. **What are the consequences of leakage with regard to control room habitability for the limiting passive leak and where and when does leak this occur and what activity level is assumed during this leakage? What is the basis for this answer?**

<u>NMC Response</u>:

For purposes of providing a limiting dose consequence due to a passive failure during recirculation post-LOCA, an evaluation was performed. The approach and assumptions used are consistent with the current licensing basis radiological design basis LOCA analysis contained in FSAR 14.3.5 and RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (May 2003).” The input used to estimate the dose consequences are delineated on Table 5.D-1.

<u>Methodology</u>

The calculation methodology described in RG 1.195, Regulatory Position 2, was used to estimate the dose to the control room. Input values needed to complete the dose estimate were taken from FSAR 14.3.5, “Radiological Consequences of a Loss of Coolant Accident.” Values chosen for parameters not specifically identified in the FSAR were based on the guidance in RG 1.195. Core activities are based on a core power level of 1549 MWt, which is the current licensed power level including calorimetric uncertainty. The thyroid dose conversion factors listed in FSAR 14.3.5, which are taken from Federal Guidance Report 11, were used. The whole body and skin dose conversion factors were taken from Federal Guidance Report 12 per RG 1.195, Regulatory Position 4.1.4. As further discussed below, decay of the activity in the sump is credited up to the point that the failure is assumed to occur. The release rate of the activity from the passive failure (Ci/min) is assumed to remain constant until the failure is isolated (i.e., removal processes such as decay is not taken into consideration). However, determination of the integrated activity in the control room does credit decay and exhaust. No other activity removal processes are credited (e.g., plate-out, hold-up, ground deposition, etc.).

<u>DBA Input and Assumptions</u>

<u>Sump Coolant Source Term</u>: Post-LOCA, 50% of the total core iodine is assumed to be in the sump coolant available for recirculation. All of the iodine released to the sump is assumed to be elemental. This assumed chemical form is consistent with the current licensing basis LOCA radiological analysis. Decay of the iodine activity in the sump coolant up to the point of the failure is credited. No credit for decay of the iodine activity in the sump is applied after the passive failure is assumed to occur. At 30 days post-accident, only significant quantities of I-131 are remaining due to the relatively short half-lives of the other isotopes of iodine. Therefore, the only activity assumed to be in the sump is I-131 based on a 200-day decay.

<u>Sump Volume</u>: Consistent with the CLB LOCA ECCS leakage dose, the amount of coolant available for recirculation is 197,000 gallons. However, it is expected that the amount of coolant available for recirculation would actually be 243,000 gallons. The increase in available sump volume is due to corrective actions taken since the licensing of the radiological LOCA analysis in 1997. At the time the LOCA analysis was under review by the NRC, it was assumed that coolant in the lower refueling cavity would not be able to drain to the “B” containment sump due to a component issue on the inlet to the cavity drain line. The cavity drain line has since been modified such that coolant in the lower refueling cavity can drain into the containment “B” sump and can be considered available for containment sump recirculation. However, to maintain consistency with the analysis, credit for the additional volume of coolant is not taken into consideration.

<u>Passive Failure Leak Rate, Occurrence, and Duration</u>: As discussed in the NMC responses to Questions 5.A and 5.B, the maximum passive failure leak rate is 50 gpm, which is postulated to occur “on the order of 200 days” following a loss of coolant accident. Therefore, the dose consequences are based on a passive failure leak rate of 50 gpm occurring at 200 days post-LOCA.

It is assumed that detection and isolation of this failure could take up to 60 hours after the onset of the failure. This assumed release duration is based on the identification of the gross leakage by loss of the suction to the decay heat removal pump without other detection methods as discussed in the NMC response to Question 5.C. However, the response to Question 5.C also states that detection of the passive failure could occur within a shift based on a gross containment sump level change. For conservatism, the more limiting detection/isolation time is assumed.

Furthermore, seat leakage past the isolated SI-850 valve is not taken into consideration because of the conservative passive failure leak rate and duration used to estimate the dose. In addition, once this failure is detected, measures would be taken to correct it.

Leakage Activity Release Fraction: Consistent with FSAR 14.3.5, the fraction of iodine in the leakage that is released to the environment is 10%. As stated in the "Methodology" section above, no credit for plate-out, hold-up, or filtration is assumed.

Release Point: For purposes of assessing the limiting dose consequence due to a passive failure during the post-LOCA recirculation mode, a passive failure occurring in the Unit 2 tendon gallery is expected to be more bounding than a passive failure inside the PAB. This is due to the fact that under the worse case assumptions, it would take longer to identify a leak in the tendon gallery than inside the PAB. The release of activity from the tendon gallery is released directly to the environment unmonitored, whereas, a release from inside the PAB is readily detectable via either PAB sump level changes, area radiation monitors or vent stack radiation monitors.

A release from the tendon gallery would be via the access hatches on El. 6.5' of the facades. The Unit 2 tendon gallery release is more limiting than the Unit 1 tendon gallery because the Unit 2 tendon gallery has an access point closer to the intake of the control room ventilation system. Of the two tendon gallery access points in the Unit 2 façade, the more limiting release point is the access point near Pipeway 4. Since both Unit 2 tendon gallery access points are within the same wind direction sector, the access release point closest to the control room intake results in larger atmospheric dispersion factors. Therefore, the bounding release point is the tendon gallery access point located in the Unit 2 façade at El. 6.5' under Pipeway 4. FSAR Figures 1.2-5 and 1.2-12 illustrate the location of the tendon gallery access points.

Atmospheric Dispersion Factors: The atmospheric dispersion factor (X/Q) associated with a release from the Unit 2 tendon gallery access hatch is 2.75E-03 sec/m^3. This dispersion factor was calculated using ARCON96 and is based on a point source, ground level release from the Unit 2 façade near Pipeway 4. The cross sectional area of the façade is used to calculate the building wake. The X/Q assumed is that value calculated for the 0-2 hour interval post-accident to provide a bounding dose estimate.

Control Room Occupancy: The control room occupancy factor is assumed to be one (1) or 100% for the duration of the passive failure. During this phase of the accident, it is expected that operators would be on 12-hour shifts. However, an occupancy factor of one (1) results in a bounding dose estimation.

Control Room Ventilation System Mode: As described in FSAR 9.8, the control room ventilation system has four modes of operation, whereby Mode 1 is the normal operating mode (outside air intake/recirculation) and Mode 4 is the emergency mode (filtered outside air intake/recirculation). It is assumed that the control room ventilation system is operating in Mode 1 and remains in Mode 1 for the duration of the passive failure. Therefore, the control room dose consequences are based on an unfiltered release. The dose consequence evaluation used an intake value of 1000 cfm, consistent with the FSAR 9.8 Mode 1 description.

Acceptance Criteria

The dose acceptance criteria are the limits delineated in 10 CFR 100.11 and 10 CFR 50, Appendix A, clarified in NUREG-0800, Section 6.4, as well as the doses documented in FSAR 14.3.5, the licensing basis radiological design basis LOCA. Further discussion of the acceptance criteria is provided in the NMC response to Question 5.F.

Dose Results: The thyroid dose to the control room operator based on the above DBA failure scenario is on the order of 0.06 rem. The whole body and skin doses are <0.0001 rem, and are therefore, negligible. This is primarily because 200 days provides a sufficient amount of decay of iodine such that a release of activity to the environment would not result in a dose of any significance with regard to control room habitability.

Design Basis Dose Consequence Licensing Basis: Based on a historical review of the licensing bases, a passive failure as posed in FSAR 6.2 to be either excessive packing/weld leakage or RHR pump seal failure, has not been assumed to occur in conjunction with the radiological design basis LOCA analysis for purposes of demonstrating compliance with 10 CFR 100 or the dose limits of GDC-19. The current licensing basis radiological accident analyses for LOCA is performed consistent with the approach used previously; namely maximum allowable containment leakage assuming failure of an emergency diesel generator resulting in one-train of containment spray and maximum allowable ECCS leakage. No additional failures are assumed during the recirculation phase.

The most significant changes made to dose analysis for the recirculation leakage have been the assumed size of the leakage from ECCS. The assumed leakage from ECCS was based on the program limits defined by the Leakage Reduction and Preventive Maintenance program, which was developed in response to NUREG-0578, Item 2.1.6.a. This change was initially communicated to the NRC in the station's final response to NUREG-0737; Item III.D.3.4 dated September 4, 1984. In addition, the filtration capability of the PAB ventilation system was eliminated from the dose analysis via Technical Specification Change Request 192, which was subsequently approved by the Commission on July 9, 1997. Although the allowed operational leakage of ECCS has increased, the methods by which the recirculation portions of ECCS are maintained and tested have not changed.

Table 5.D-1

Input Assumptions Used to Estimate Control Room Operator Dose Due to a Passive Failure in the Unit 2 Tendon Gallery Post-LOCA

Input	Value
Core Power (includes calorimetric uncertainty)	1549 MWTh
Total Core Iodine	4.13E+07 Ci
Fraction of Total Core Iodine in the Sump	0.50
Sump Volume	197,000 gal
Passive Failure Leak Rate	50 gpm
Iodine Re-evolution Release Fraction	10%
Duration of Passive Failure Leak	60 hr
Tendon Gallery –CR Atmospheric Dispersion Factor	2.8E-03 sec/m^3 (0-2 hr)
Height of Lower Instrumentation	10 m
Control Room Parameters	
Breathing Rate	3.5E-04 m^3/sec
Occupancy	1
Control Room Volume	65,243 ft^3
Outside Air Intake	1000 cfm
Filtered Outside Air Intake	0 cfm

E. What are the consequences of leakage with regard to offsite dose for the limiting passive leak and where and when does this leak occur and what activity level is assumed during this leakage? What is the basis for this answer?

NMC Response:

The estimation of the offsite dose consequences due to the limiting passive leak (i.e., passive failure) follows the basis for the control room consequences as documented in the response to Question 5.D with two exceptions: One with regard to the atmospheric dispersion factors (X/Q), and the second with regard to assumed dose duration for the site boundary. Since the atmospheric dispersion factors are not release point specific, there is no difference in offsite dose due to a passive failure in either the Unit 1 or Unit 2 tendon galleries or PAB. The offsite X/Qs represent an overall site dispersion of a release. Therefore, the 0-2 hour atmospheric dispersion factor for the site boundary (5.0E-04 sec/ m^3) and the 0-2 hr atmospheric dispersion factor for the low population zone (3.0E-05 sec/m^3) from the licensing basis LOCA radiological consequence analysis (FSAR 14.3.5) are used. The site boundary doses are calculated for the first two hours of the release, whereas, the low population zone doses are calculated for the duration of the release. Other source term assumptions and their bases as discussed in the Response to Question 5d remain the same for determining dose consequences to the offsite. Similarly, the method presented in RG 1.195, Section 2, was also used to estimate the offsite doses. Only decay up to the point of the failure is credited for reducing the source term in the containment sump.

Dose to the Offsite Following a Passive Failure at 200 days Post-LOCA: The thyroid and whole body doses to the site boundary and low population zone based on the DBA passive failure scenario that occurs at 200 day post-LOCA are less than 0.001 rem; therefore, negligible. This is primarily due the fact that 200 days provides a sufficient amount of decay of iodine such that any release of activity to the environment would not result in a dose of any significance to the offsite. Acceptability of the results is discussed in the response to Question 5.F.

F. Are the consequential radiation exposures within calculated results and regulatory limits? What is the basis for this answer?

NMC Response:

The passive failure dose consequences, as well as, the current licensing basis radiological consequences documented in FSAR 14.3.5 for the control room habitability and offsite consequences and regulatory limits are provided in the Table 5.F-1 below. The use of the symbol "-" indicates no dose limit identified or that a dose is not required to be calculated.

The current licensing basis for PBNP control room habitability includes a factor of ten (10) dose reduction credit for the ingestion of potassium iodide (KI). The control room thyroid doses documented below include this credit. Credit for the ingestion of potassium iodide (KI) was not applied to the control room passive failure thyroid dose consequence.

Based on the table, it is seen that the doses due to the passive failure are within regulatory limits and bounded by the current licensing basis radiological design basis accident analysis. The passive failure doses calculated for the control room are conservative since control room filtration is not credited, 100% occupancy is assumed and the worst case meteorological conditions are applied for the duration of the accident.

Table 5.F-1 Dose Consequences

Location/Release Path	Thyroid (rem)	Whole Body (rem)	Skin (rem)
Regulatory Limit - CR	30	5	75*
CLB CR - Total	29.27	1.37	43.18
Containment Leakage	18.60	1.366	43.14
ECCS Leakage	10.67	0.004	0.04
Passive Failure	0.06	4E-07	2E-05
Regulatory Limit - SB	300	25	-
SB - Total	190.42	3.48	-
Containment Leakage	133.3	3.24	-
ECCS Leakage	57.12	0.24	-
Passive Failure	4E-04	6E-08	
Regulatory Limit - LPZ	300	25	-
LPZ - Total	61.37	0.51	-

Table 5.F-1 Dose Consequences			
Location/Release Path	Thyroid (rem)	Whole Body (rem)	Skin (rem)
Containment Leakage	24.37	0.45	-
ECCS Leakage	37.0	0.06	-
Passive Failure	6E-04	1E-07	

*As defined in SRP 6.4, the skin dose limit is 30 rem, unless the licensee commits to use of protective clothing and goggles during a severe radiation release. Then the unprotected skin dose limit is not to exceed 75 rem. PBNP committed to maintain protective clothing and goggles in the control room in response to NUREG-0737, Item III.D.3.4, on February 23, 1981, and reconfirmed in letter dated September 4, 1984..

G. What are the consequences of passive leakage and isolation capabilities with respect to ECCS functions (e.g., preservation of containment sump inventory to support post LOCA recirculation)? What is the basis for this answer?

NMC Response:

A passive leak in the ECCS outside containment will be detected and isolated prior to the loss of containment sump inventory to the extent that core cooling capabilities will not be challenged.

As discussed in the response to Question 5.B, there is reasonable assurance that a postulated leak of 50 gpm in the containment sump suction line would be detected and isolated prior to loss of pump suction as a result of loss of containment sump inventory. This is regardless of the leak location, whether in the PAB or the tendon gallery.

Flooding in the tendon gallery or PAB as a result of a passive failure in the RHR suction line will not prevent the ability of the ECCS system to perform its safety function for core cooling. Equipment lost in flooding of the tendon gallery would consist of the tendon gallery sump pump. Loss of this nonsafety-related pump has little effect on the ability to detect and isolate a passive failure as demonstrated in the NMC response to Question 5.B. Flooding in the PAB will likely result in the loss of one RHR pump prior to leak isolation but the other train RHR pump would be available to maintain core cooling. Makeup water to the containment sump is available and is procedurally directed as a contingency if there is a loss of reactor injection flow as a result of inadequate containment sump performance.

H. Are the SI-850 valves credited with isolating a passive leak? If so, is this a safety-related function? If not, explain. What is the basis for this answer?

NMC Response:

The SI-850A(B) valves perform a safety-related function to isolate a passive failure in the containment sump recirculation line to prevent gross diversion of containment sump inventory. The SI-850A(B) valves would be shut to support the following post-accident functions following a credible leak in the containment recirculation line:

- Maintain Sump “B” inventory
- Protect the RHR system and pumps from flooding

The shut safety-related function is discussed in FSAR Chapter 6.2.2 where it states:

“Each recirculation sump line has two remotely operated valves. The first valve is located adjacent to the end of the pipe in the containment such that the line inside the containment can be isolated in the event of a passive failure.”

In accordance with 10 CFR 50.2, the ability of the SI 850A(B) valves to isolate a passive failure is classified as a safety-related function. The ability of the SI 850A(B) valves to isolate a passive failure supports Criteria 2 and 3 for a safety-related component. Shutting these valves to isolate a passive failure prevents the gross diversion of containment sump inventory and ensures that at least one redundant train of long-term core cooling remains operable throughout the post-accident phase. Long-term decay heat removal is essential to maintain the plant in a safe shutdown condition and to ensure offsite doses are maintained within the limits of 10 CFR 100 and control room doses are within the limits of 10 CFR 50 Appendix A, GDC-19.

PBNP is designed to withstand the maximum credible leakage of 50 gpm from the containment sump recirculation lines and the RHR system without a loss of capability to shut down the reactor and maintain it in a safe shutdown condition. PBNP did not previously consider that the SI-850A(B) valves performed a safety-related function in the shut position. A corrective action program document was initiated in response to NRC inspection questions during the November 2005 inspection.

Although not previously evaluated against acceptance criteria, previous test data taken on these valves in the shut direction for trending purposes were within the bounds of the ASME Code-required acceptance bands. This demonstrates that the valves are fully capable of performing a safety-related function to shut. Since it was determined that the SI-850A(B) valves perform a safety function to shut, the PBNP IST program document has been updated to reflect this function and the associated IST implementing procedures (IT 40 and 45) have been revised to include acceptance criteria for shutting the valves. These revised procedures have been implemented. The four SI-850A(B) valves have had satisfactory test results.

Please refer to the NMC response to Question 5.J related to isolation of a containment sump recirculation line following a passive failure for dose considerations.

I. If the SI-850 valves are credited with isolating a passive leak, explain how much this valve will continue to leak after closure and how this leak rate was determined. If this leak rate has not been measured, explain what a limiting leak rate would be and your basis for this leak rate. What is the basis for this answer?

NMC Response:

The SI-850A(B) valves are credited with isolating a passive leak in the containment sump recirculation lines. PBNP does not perform a seat leakage test on the valves in the direction of the containment sump to the recirculation lines. Based upon the design of the valves and their operating conditions, PBNP expects the valves to limit a passive pressure boundary failure sufficiently preventing a gross diversion of water from the containment sump.

Although not previously credited, NMC has determined that the SI-850A(B) valves perform a safety-related function to shut and isolate a passive failure. PBNP does not perform a seat leakage test on the SI-850 valves in the direction of the containment to the containment recirculation line. Therefore, PBNP does not have qualified seat leakage data on these valves.

The safety-related function to shut is to isolate a passive failure in the RHR containment sump recirculation line. The design leakage rate of the passive failure is bounded by a 50 gpm leak. Shutting the SI-850 valves will reduce the bounding leakage rate of 50 gpm. A specific maximum seat leakage rate is not required to reach a manageable leak rate with respect to maintaining the decay heat removal function, as PBNP is designed to withstand the bounding 50 gpm leak rate. Based on this, the SI-850 valves were not intended to meet the requirements to be classified as Category A valves per ASME OM Code, Paragraph ISTC 1.4(a), which states:

"Category A – valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required functions."

The valves are, however, required to prevent gross diversion of water through a passive failure in the containment sump recirculation line and are classified as Category B valves. Per ASME OM Code, Table ISTC 3.6-1, "In-service Test Requirements," seat leakage testing is not required for Category B valves.

While the PBNP licensing/design basis limits the leakage from this passive failure in the RHR to less than 50 gpm, the design of the SI-850 valves is expected to significantly reduce the leakage rate when they are shut. The SI-850 valves are equipped with a resilient (soft) seat. Resilient seats are used to accomplish good seating performance with much lower contact force than is required in metal-to-metal seats. In the case of the SI-850 valves, the resilient seat is formed by an O-ring and provides the primary seating seal with the metal-to-metal closure acting as a secondary seal. . Based on ANSI B 16-104, American National Standard for Control Valve Seat Leakage, the allowable seat leakage for a valve with the design of the SI-850 valves (Class VI) would be approximately 15 ml/min at maximum rated differential pressure. While not an element of the PBNP licensing basis, ANSI B 16-104 is an industry standard used to determine expected leakage of resilient seals. A review of the forces on the containment sump "B" isolation valves concluded that adequate sealing forces are applied for the O-ring to provide and adequate seal as ascertained in Engineering Evaluation 2006-0003.

Although the soft-seated design of the SI-850 valves would be expected to control seat leakage to a very nominal rate, as they are classified as Category B valves per ASME OM Code, no seat leakage testing is performed on these valves to quantify this leakage rate. Based on the above discussion, PBNP would expect a shut SI-850A(B) valve to prevent the gross diversion of water from the containment sump through the containment sump recirculation line.

J. Was the continued leakage past the shut SI-850 valve considered in calculation of control room dose, off-site dose or preservation of containment sump inventory? If not, explain. What is the basis for this answer?

<u>NMC Response</u>:

As discussed in response to Question 5.D (calculation of control room dose), seat leakage past the isolated sump suction isolation valve, SI-850, was not taken into consideration as a result of the conservative passive failure leak rate and duration used to estimate the dose consequences. Since the calculation of offsite dose (response to Question 5E) used the same methodology as the control room dose calculation, the exclusion of seat leakage has the same basis.

As stated in the NMC response to Question 5.G, makeup to the containment sump is available and is procedurally directed as a contingency if containment sump performance is identified as a concern. Once a passive failure is identified and isolated, leakage past the shut containment sump isolation valve is negligible (refer to NMC response to Question 5.I).

ENCLOSURE 2

ATTACHMENT 1

The following attachments to Enclosure 2 are provided to assist in the review of the NMC response to the RAI:

Attachment	Description
1 Pages 1-3	Unit 1 Delaminating Qualified Coatings List
Pages 4-7	Unit 1 Containment Elevations Showing Degraded Or Nonconforming Coatings
2 Pages 1-3	Unit 2 Delaminating Qualified Coatings List
Pages 4-7	Unit 2 Containment Elevations Showing Degraded Or Nonconforming Coatings
3	1(2)SI-850A(B) SIS Drains Elevation Sketch
4	1(2)SI-850A(B), SIS Drains Plan
5	1(2)SI-850A(B) Valve

ENCLOSURE 2

ATTACHMENT 1

Unit 1 Delaminating Qualified Coatings

Map #	Location	Area (ft^2)	Description
Reactor Cavity			
1	Reactor cavity	10	Delaminated and cracking coatings on walls
Subtotal		**10**	
8' Elevation General Area			
2	Az 45, El 10	2	on ICW
3	Az 85	3	ICW, delamination
4	Az 90, El 20	8	Penetration 28, light rust
5	Az 140	12	WTUP on 3 columns and ceiling – qualified by adhesion test. Some flaking
6	Az 150	0.5	By LP, support peeling at bolts
7	Az 170	8	Flaking topcoat on 2 columns. In general, steel embeds have red or zinc primer with white topcoat. Concrete has green surface (up to 1/8-inch thick), with white, gray, and white intermediate and top coats. Most areas appear tight, with a few areas having delaminating topcoat
8	Az 200	8	Delaminating topcoat on column
9	Az 200	1	On column, degraded coating over unprepared steel embed
10	Az205	3	Inner concrete column, total failure of coating; no adhesion of base coat
11	Az 235	6	ICW, support with poor application over red primer, loose and chipping off
12	Az 245, El 14	0.5	LP, topcoat loose and chalky
13	Az 270	3	In cubicle opening, delaminated concrete coating
14	Az 310	3	On column near LP, horizontal pipe member of support, poorly done over red primer
15	Az 357	1	Keyway wall, checking and delamination.
Subtotal		**59**	
A Steam Generator Cubicle			
16	1st Level	6	SG support struts. – Most loose coatings removed.
17	1st Level	16	Northeast corner column, cracked and delaminated
18	3rd Level	2	Northwest wall.
19	3rd Level	10	Northeast wall.
20	3rd Level	20	East wall, degraded coatings on a penetration through the east wall towards the reactor.
21	4th Level	10	South wall
71	5th Level	40	East wall
72	5th Level	30	South wall
73	5th Level	10	Floor, south
74	5th Level	40	West wall
Subtotal		**184**	

ATTACHMENT 1

Unit 1 Delaminating Qualified Coatings

Map #	Location	Area (ft^2)	Description
A Reactor Coolant Pump Cubicle			
22	1st Level	5	On the RCP support struts. – most loose coatings removed.
23	3rd Level	8	South wall
24	4th Level	8	South wall cracking and delamination
25	5th Level	4	Top of the Upper Oil Cooler and pipe
26	5th Level	10	Slab joints/ledge above, delaminating
27	5th Level	18	South wall, a steel structure for HVAC – loosely adherent coating, easily removed
28	5th Level	5	Southwest wall, cracking, delaminating, WTUP.
	Subtotal	**58**	
B Steam Generator Cubicle			
29	Snubber Level	25	North wall, delaminating
30	Snubber Level	25	East wall, delaminating
31	Snubber Level	50	South wall, delaminating – bad surface prep
32	Snubber Level	25	West wall, delaminating
	Subtotal	**125**	
B Reactor Coolant Pump Cubicle			
33	1st Level	3	North, scratch, column delaminated
34	1st Level	1	North opening, large blister on top part
35	3rd Level	10	Oil pipes, degraded, poor surface prep (shiny or mill scale)
36	3rd Level	4	South wall, cracking and delamination
37	3rd Level	8	North side, cracked, delaminating, embed
38	3rd Level	14	Northeast, cracked and delaminating
39	4th Level	12	North wall, cracks and delamination
40	4th Level	6	East wall, cracks and delamination
41	4th Level	6	South wall, cracks and delamination
42	5th Level	6	Northeast wall, cracking and delamination
43	5th Level	6	South wall, cracking and delamination
44	5th Level	6	West wall, cracking and delamination
	Subtotal	**82**	
Pressurizer Cubicle			
45	Top level	16	North wall, delaminating. Bad surface prep
46	Top level	12	East wall, orange-tan touchup, checking, delamination
47	Top level	6	Southeast wall, delamination, checking and cracking
48	Top level	10	South wall, delaminating and cracking
49	Top level	6	West wall, checking and delamination
50	Base	42	Floor 35% abraded area about 120 ft^2. Condition not good
51	Cubbyhole	19	Walls, ceiling
	Subtotal	**111**	

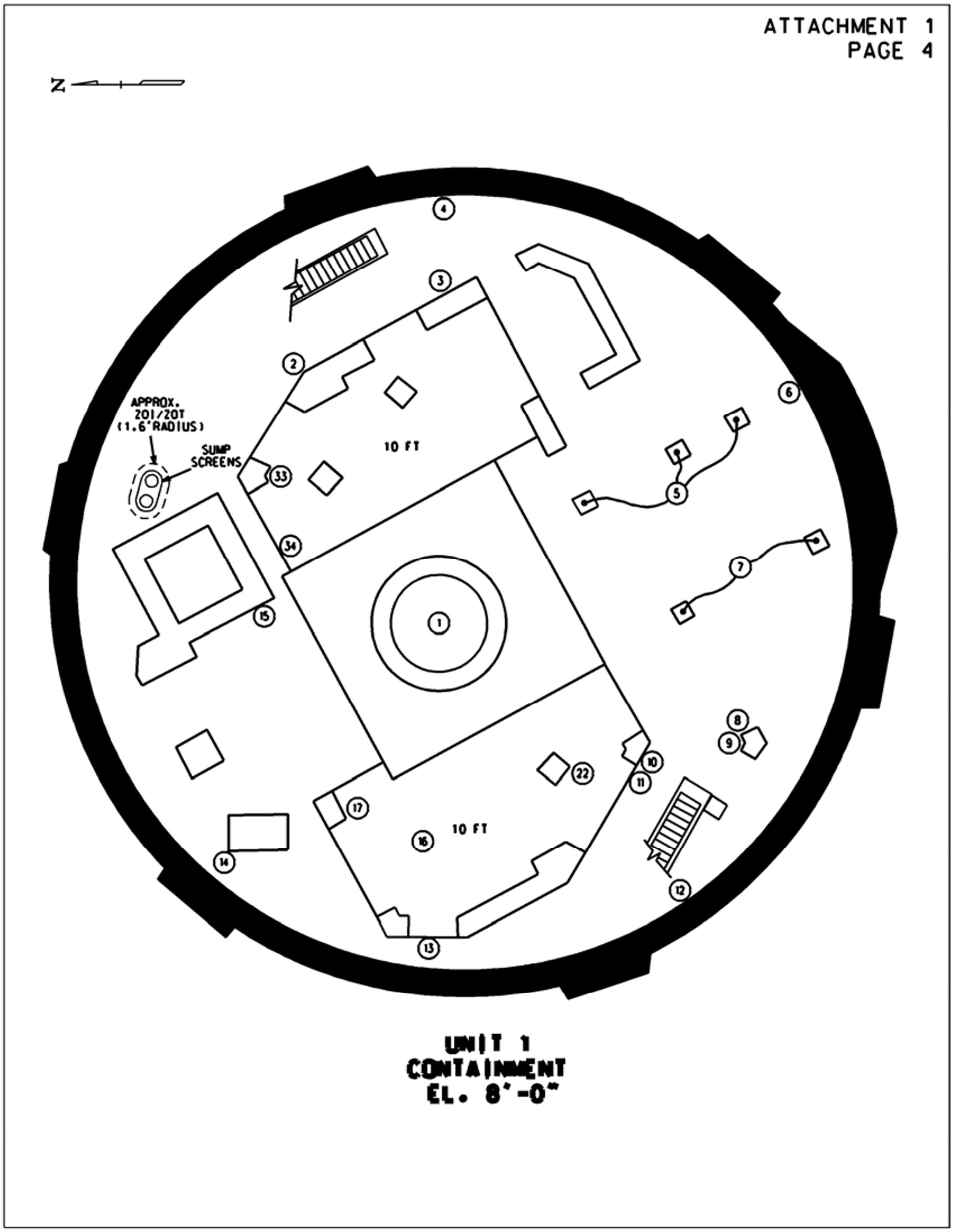
ATTACHMENT 1

Unit 1 Delaminating Qualified Coatings

Map #	Location	Area (ft^2)	Description
21' Elevation General Area			
52	Az 45	1	Delaminating coating on support column for 1W1C1 – bad surface prep
53	Az 92, El 22	8	LP, penetration 34, light rust at welds
54	Az 130	1	By LP, steel column with delaminated topcoat, zinc primer intact, no rust
55	Az 210 to 249	100	ICW, cracked/delaminated coating
56	Az 310	50	ICW, cracked/delaminated coatings & WTUP
	Subtotal	**160**	
46' Elevation General Area			
57	Az 40	2	Penetration 27 – medium rust
58	Az 42, El 49	3	On penetration through the ICW toward the reactor cavity (NE-133 or M-300-7-1)
59	Az 49, El 60	10	delaminated coating on the LP, no rust
60	Az 150	4	By LP, steel column, delaminating coating, applied over dirt or grease?
61	Az 230,	4	Halfway downstairs, ICW chipping, grout holes
62	Az 245	10	Cracked and peeling on inner wall
63	Az 259	10	Cracked and peeling coating on the inner concrete wall, especially by the embeds
	Subtotal	**43**	
66' Elevation General Area			
64	Az 66	10	Cracks and peeling coating on the inner containment wall
65	Az 90	2	ICW, up high, cracked and delaminated
66	Az 105-135	120	400 ft^2 area, floor coating between the hatch open area and the inner concrete walls is 30% cracked and abraded, coating is not tight and chips easily – CAP029629, WO 0212790
67	Az 115	1	ICW, delamination
68	Az 120	10	ICW, cracked and peeling paint, WTUP, orange touchup on the southwest wall of 1HX1B
69	Az 265	1	ICW delaminated
70	Az 275	20	ICW, delaminating and peeling coating, especially around the embeds
	Subtotal	**164**	
	Total	**996**	

ATTACHMENT 1

Unit 1 Containment Elevations Showing Degraded Or Nonconforming Coatings



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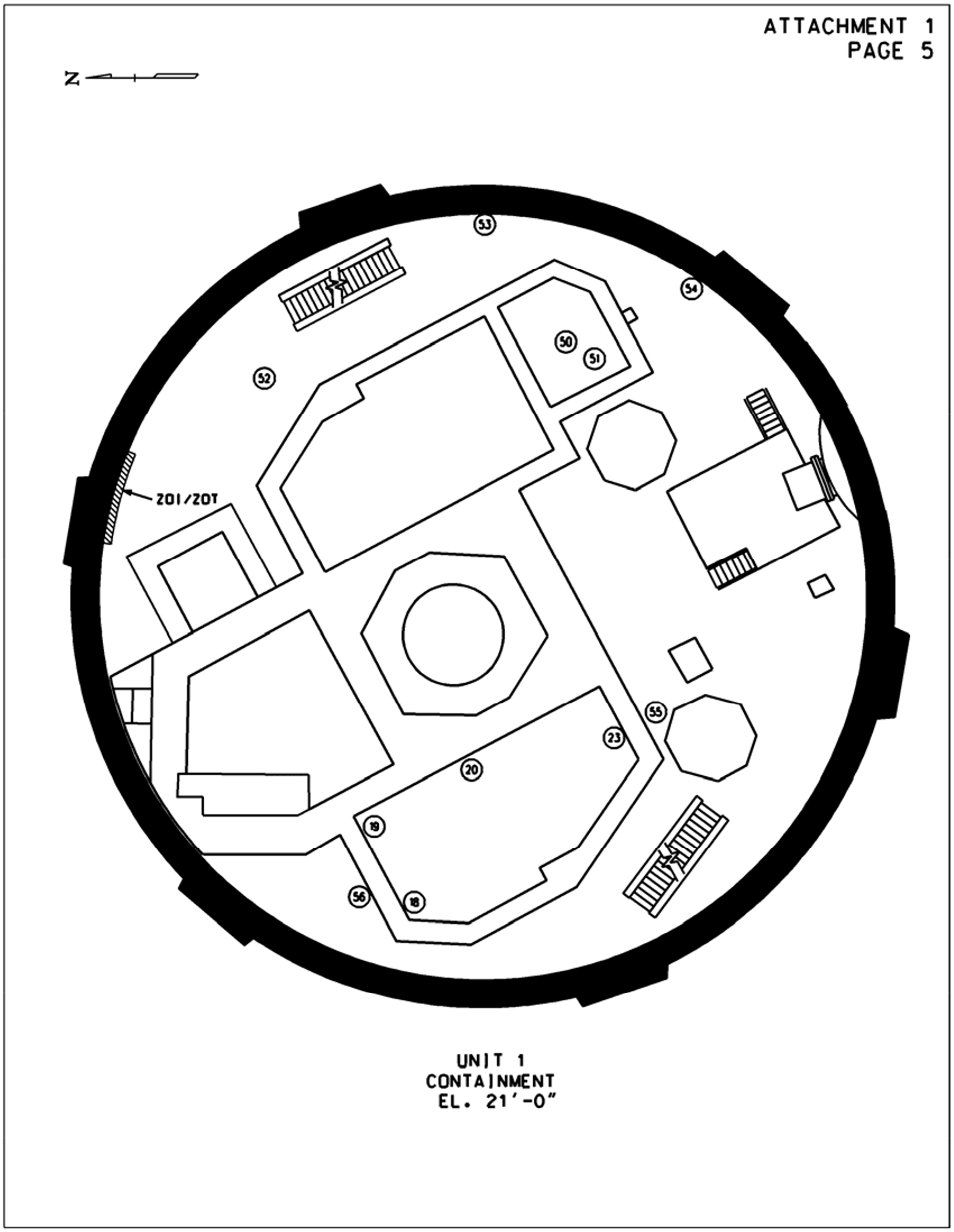
ENCLOSURE 2

ATTACHMENT 1

Unit 1 Containment Elevations Showing Degraded Or Nonconforming Coatings



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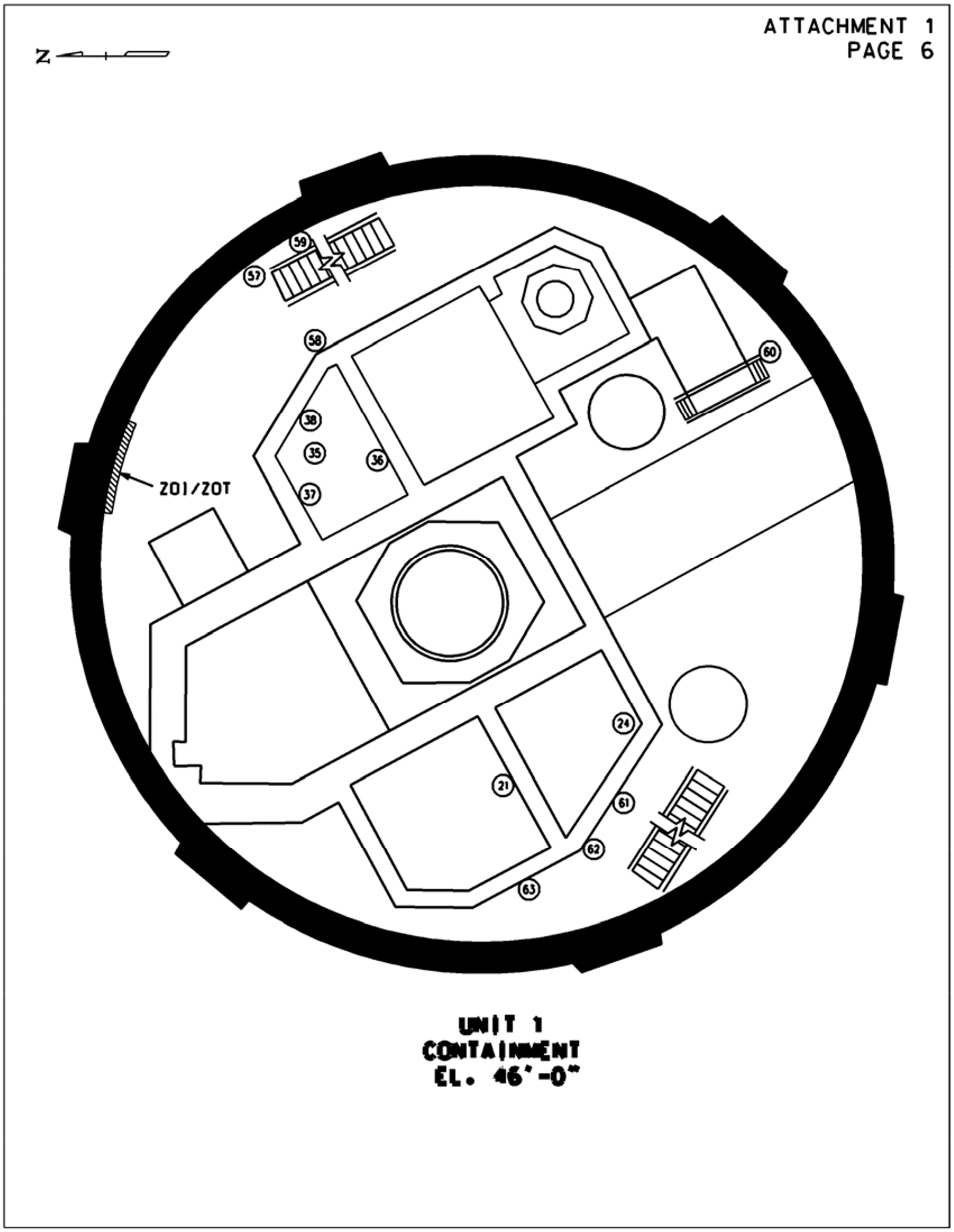
ENCLOSURE 2

ATTACHMENT 1

Unit 1 Containment Elevations Showing Degraded Or Nonconforming Coatings



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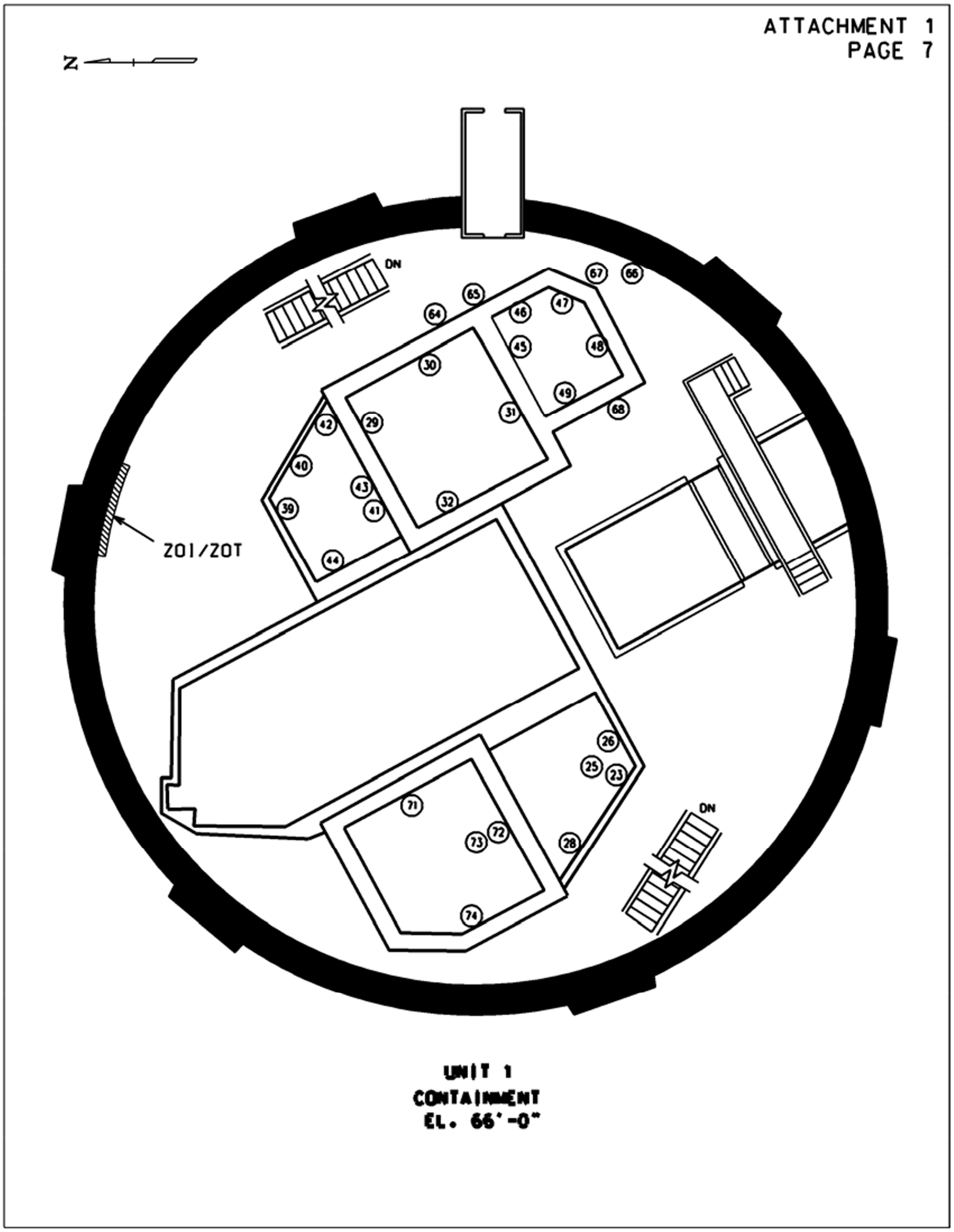
ENCLOSURE 2

ATTACHMENT 1

Unit 1 Containment Elevations Showing Degraded Or Nonconforming Coatings



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ATTACHMENT 2

Unit 2 Delaminating Qualified Coatings

Map #	Location	Area (ft^2)	Description
Keyway			
1	Access shaft	40	LP, rust dripping and loose joint material at the horizontal transition joint, concrete to the LP
2	Floor (entire)	190	Approximate 380 sf area, 50% delaminated
3	Floor (entire)	100	LP, debris strewn. Require cleaning for proper inspection
4	Base, Sump A	100	LP, standing water with dirt/debris, condition of the floor was inaccessible. No obvious evidence of rust on floor.
5	Tunnel	100	LP, southwest, floor, debris pile at the kick plate separating the tunnel from the access shaft. The kick plate is not sealed. Debris has paint chips in it.
6	Reactor room	30	Tunnel opening, Southeast wall. Concrete top coat delaminating
7	Reactor room	30	Concrete wall. Delaminating concrete coating at a construction joint
	Subtotal	**590**	
8' Elevation, General Area			
8	Az 149, El 20	4	LP, service water penetration P08, light rust
9	Az 240, El 16	1	East face of Sump A shaft, delamination over steel embed
10	Az 270	15	Entry to SG cubicle, delamination of white touch-up.
	Subtotal	**20**	
A Steam Generator Cubicle			
11	Entryway, El 12	20	East of the East wall, tape residue and degraded concrete coating on the Reactor wall and the East wall
12	Base	185	Walls, along perimeter Delaminating and cracked coating distributed on all walls
13	2nd L	70	Walls, cracked and delaminated coating
14	3rd L	270	Walls, cracked and delaminating coating and touch-up
	Subtotal	**545**	
A Reactor Coolant Pump Cubicle			
15	Base, El 20	110	Walls, along a perimeter. Delaminating and cracked coating distributed on all walls
16	2nd L	70	Walls concrete coating, delaminated, and white touch up
17	3rd L	210	All walls, distributed. Concrete coating delaminated, and touch up – no adhesion of base coat
18	3rd L	55	RCP, top portion of the bottom half. Degraded coating, easily removed, apparently not insulated
19	3rd L	10	RCP, bottom flange of the top half, flange perimeter is degraded
20	3rd L	50	East wall, cracks and delamination
21	4th L	270	Walls, concrete coating, delaminated, and touch up
	Subtotal	**775**	

ATTACHMENT 2

Unit 2 Delaminating Qualified Coatings

Map #	Location	Area (ft^2)	Description
		B Steam Generator Cubicle	
22	Base	165	All Walls (not as bad as “A” cubicles)
23	Base	2	2 large columns with hairline cracking and delamination at top corners
24	2nd Level	110	All walls, delamination
25	3rd Level	195	East Wall
26	3rd Level	60	South Wall
27	3rd Level	10	West Wall
28	4th Level	200	North wall coating in very poor condition
29	4th Level	500	East and notch wall coating in very poor condition
30	4th Level	70	South & West walls
	Subtotal	**1312**	
	B Reactor Coolant Pump Cubicle		
31	Base	2	2 large columns with hairline cracking and delamination at top corners
32	2nd Level	105	All walls, delamination
33	4th Level	60	All walls
34	Top L	35	Concrete wall coating, delaminated top coat
	Subtotal	**202**	
		Pressurizer Cubicle	
35	Top Level	3	Spalled concrete and degraded coating 6 to 7 feet below access opening
36	Top Level	1	Degraded coating on wall at top of ladder
37	Mid Level	20	Wall coating, delaminating Small platforms
38	Bottom Level	50	Wall touch up, grout holes and delaminations
39	Bottom Level	3	Fire damage near door to RCP Cubicle
	Subtotal	**77**	
		21’ Elevation General Area	
40	Az 0 to 30	10	ICW, cracked and delaminating coating
41	Az 12	35	Head laydown stand. Steel coating is severely degraded. Concrete coating appears tight
67	Az 90, El 29	2	ICW, (north wall), 4 sf grout holes & 2 sf degraded concrete
42	Az 145, El 28	10	LP, 6 of the 36 penetrations have light to medium rust and/or degraded coating. CAP064095, WOs 0501976, 0501977, and 0501978
43	Az 148 to 153	5	LP, SW pipe through 2CPP-45&46 has heavy rust bleeding through the coating (originally ~18 sf, 11 sf removed)
44	Az 250, El 25	20	Wall, coating, delaminating
45	Az 260, El 29	10	Wall coating, delaminating

ATTACHMENT 2

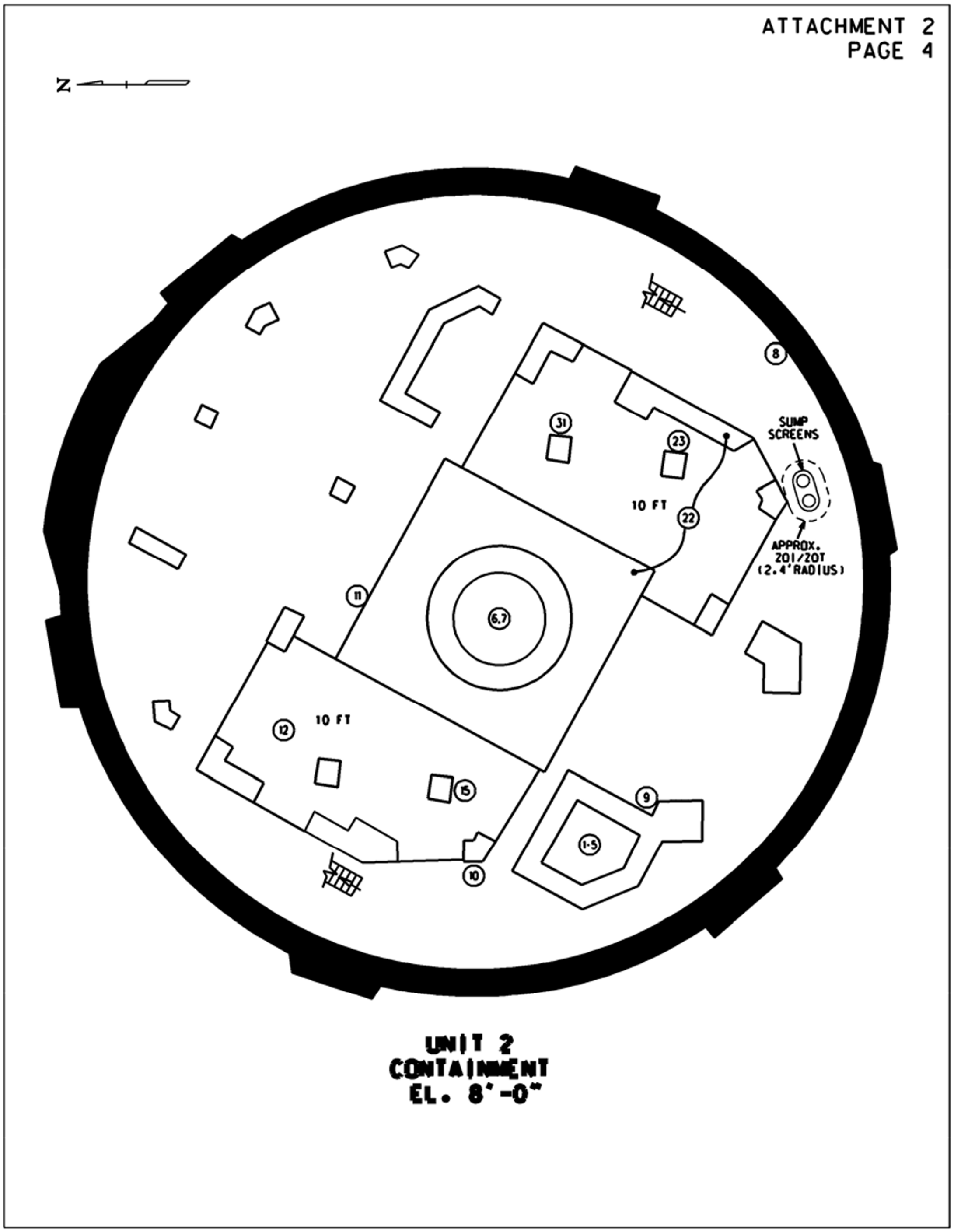
Unit 2 Delaminating Qualified Coatings

Map #	Location	Area (ft^2)	Description
21' Elevation General Area (continued)			
46	Az 266, El 26	15	By LP, cavity cooling valve area. Most of the coating appears tight, some areas have rust bleeding through coating CAP051481, WO 0309879
	Subtotal	**107**	
46' Elevation General Area			
47	Az 0 to 10	6	Floor area abraded and delaminating
48	Az 110, El 60	10	ICW, face of B RCP East wall: Wall, grout holes, degraded coating, degraded supports
49	Az 115	20	By LP, floor delaminating. Failure of the concrete itself, not just the coating
50	Az 135, El 58	1	ICW, delaminated concrete top coat
51	Az 148, El 63	2	ICW, floor to ceiling line, degraded (light rust) floor penetration
52	Az 158, El 51	10	ICW, face of B SG Southeast wall: Degraded coating on 2AC12, 2AC13 & grout holes
53	Az 225 to 269	60	Floor area 305 sf, 20% abraded – POOR ADHESION
54	Az 255, El 46	4	Floor and steel surrounding insulated HB-1 riser; poor adhesion, chipping off
55	Az 300 to 320	10	ICW, long horizontal crack with delamination
56	Az 309, El 51	6	ICW, delaminating concrete coating, with tiny cracks
57	Az 310, El 58	10	ICW, degraded concrete coating
58	Az 310, El 48	2	ICW, 2 large circular areas delaminating
	Subtotal	**141**	
66' Elevation, General Area			
59	Az 0, El 74	1	ICW
60	Az 70, El 99	10	LP, penetration V02. Control equipment
61	Az 89, El 103	2	By LP, Crane access platform, Crane rail girder support by platform
62	Az 90, El 99	10	LP, penetration V01, control, (1AT402)
63	Az 240	20	East Wall of "A" SG
64	Az 293, El 97	120	Top of South A SG wall, delaminating concrete
65	Az 325	2	ICW
66	Az 300, El 115	6	LP, few large blisters over globs of grease or dirt
	Subtotal	**171**	
	Total	**3940**	

ENCLOSURE 2

ATTACHMENT 2

Unit 2 Containment Elevations Showing Degraded Or Nonconforming Coatings



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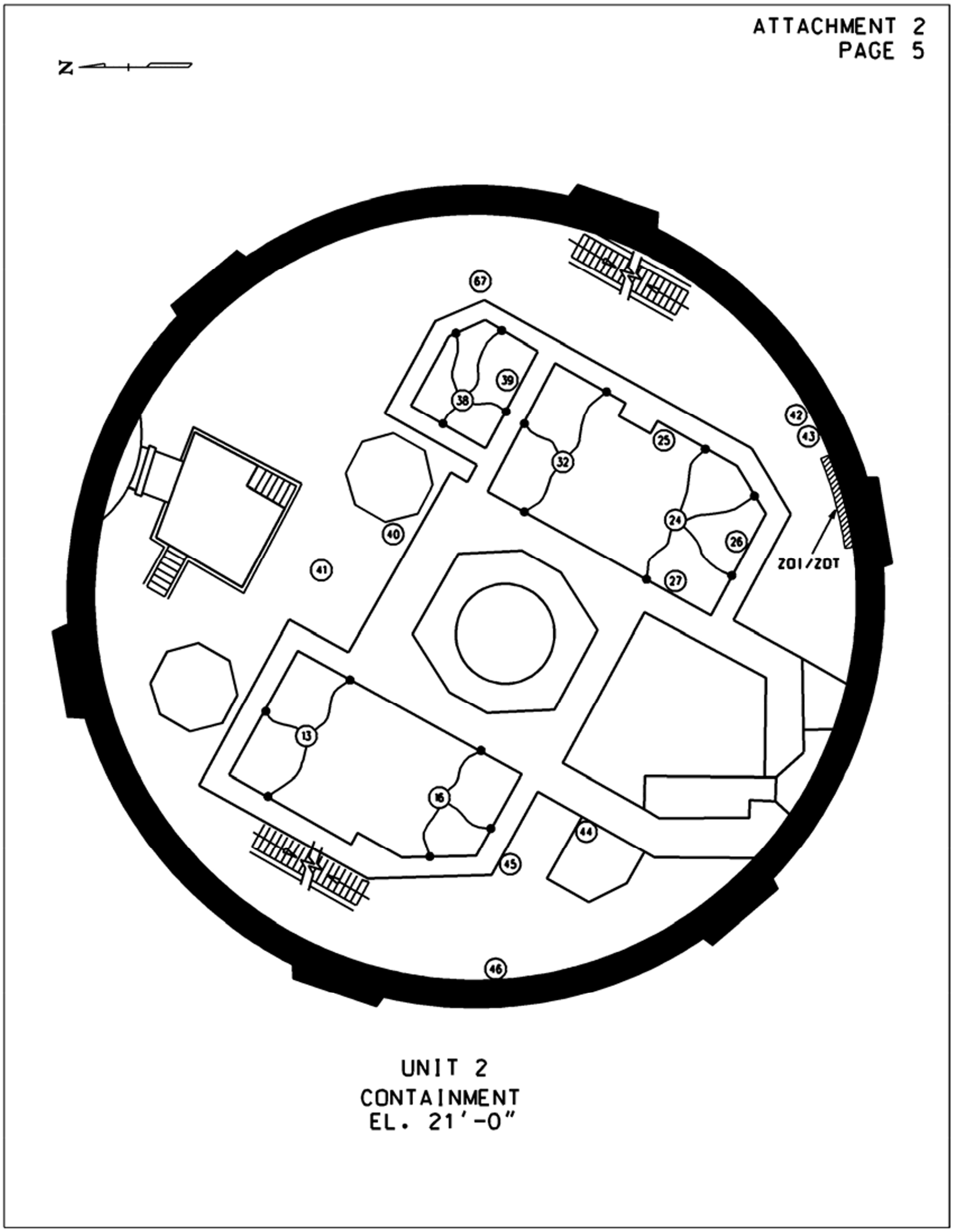
ENCLOSURE 2

ATTACHMENT 2

Unit 2 Containment Elevations Showing Degraded Or Nonconforming Coatings



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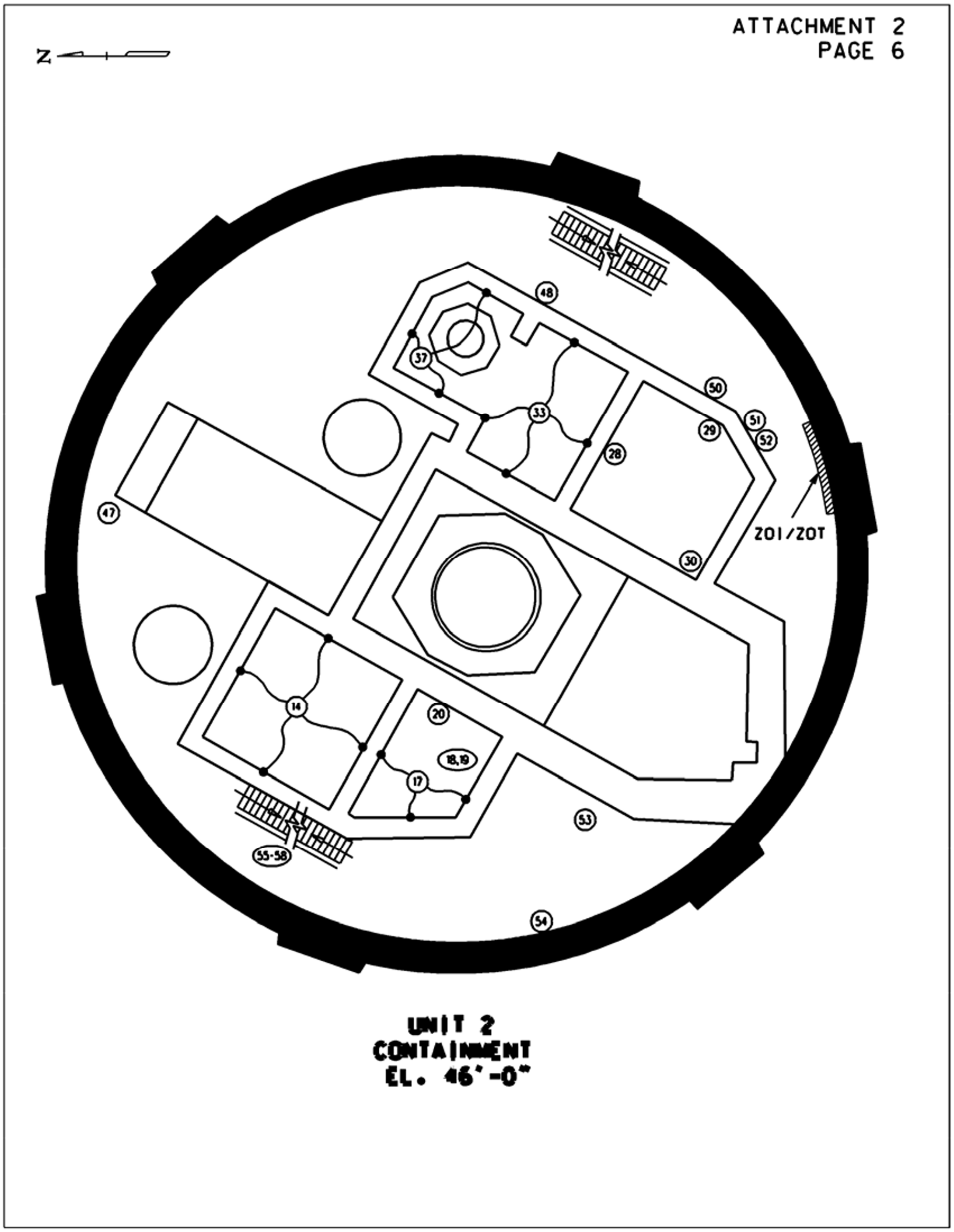
ATTACHMENT 2

Unit 2 Containment Elevations Showing Degraded Or Nonconforming Coatings



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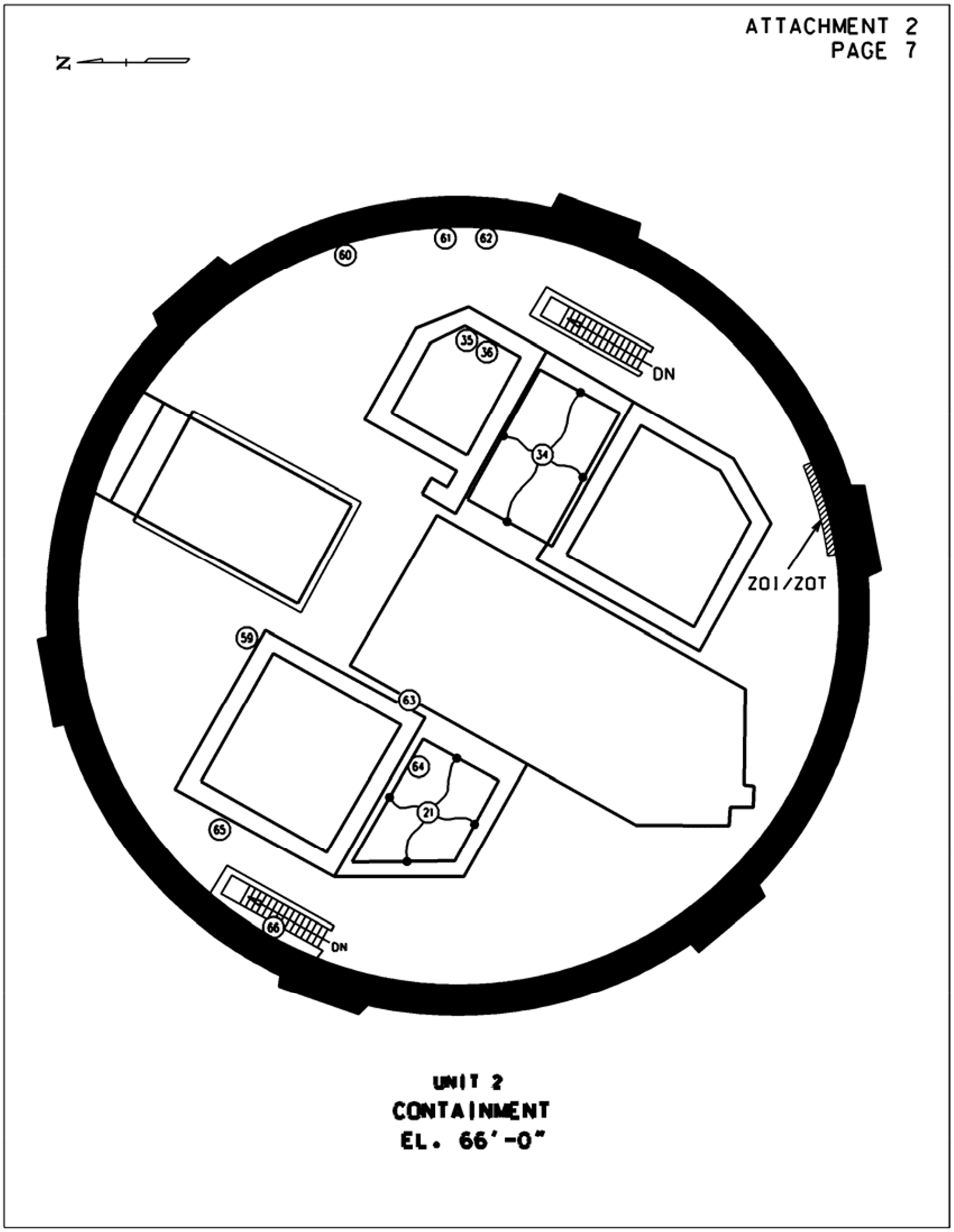
ATTACHMENT 2

Unit 2 Containment Elevations Showing Degraded Or Nonconforming Coatings



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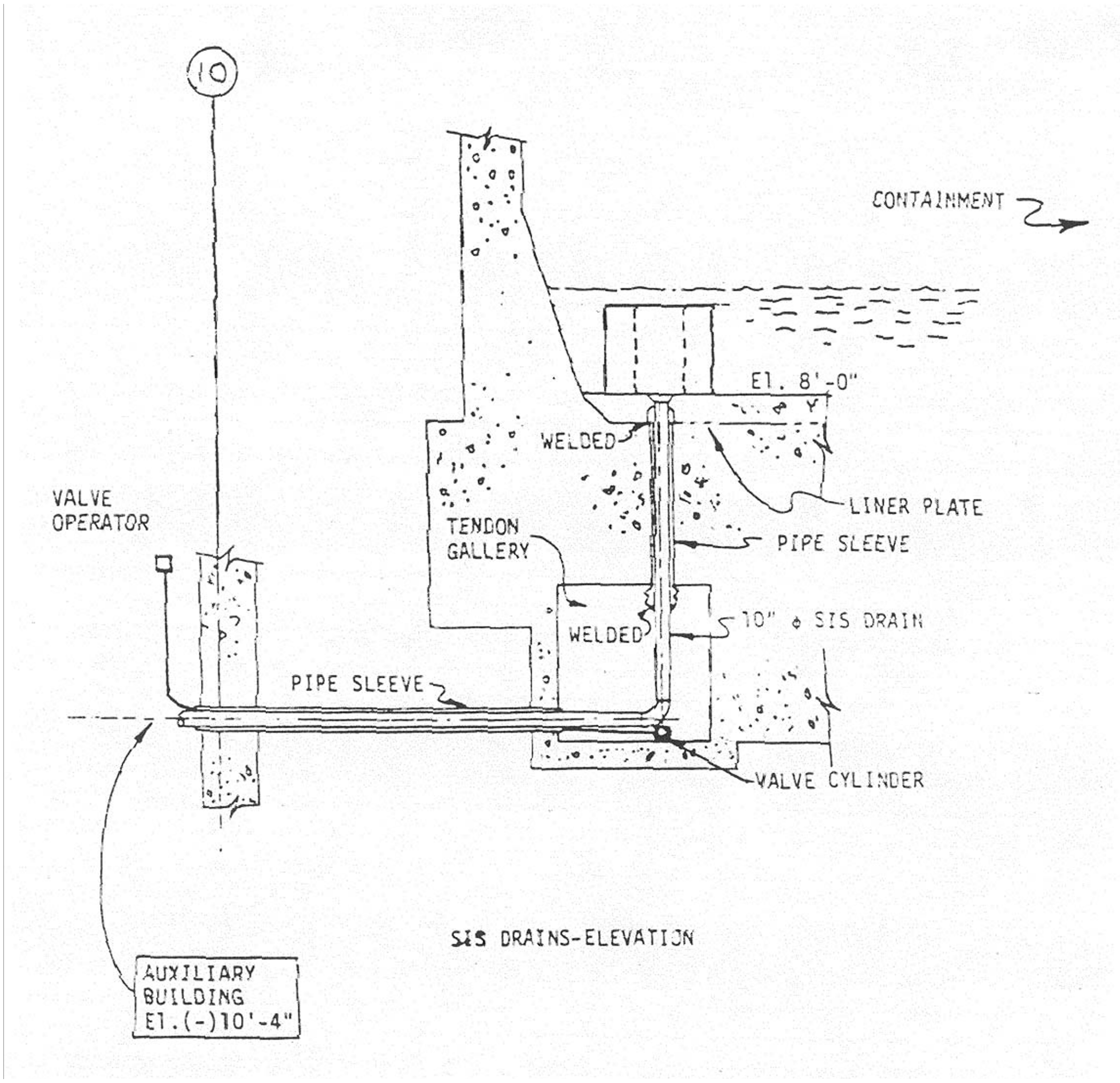
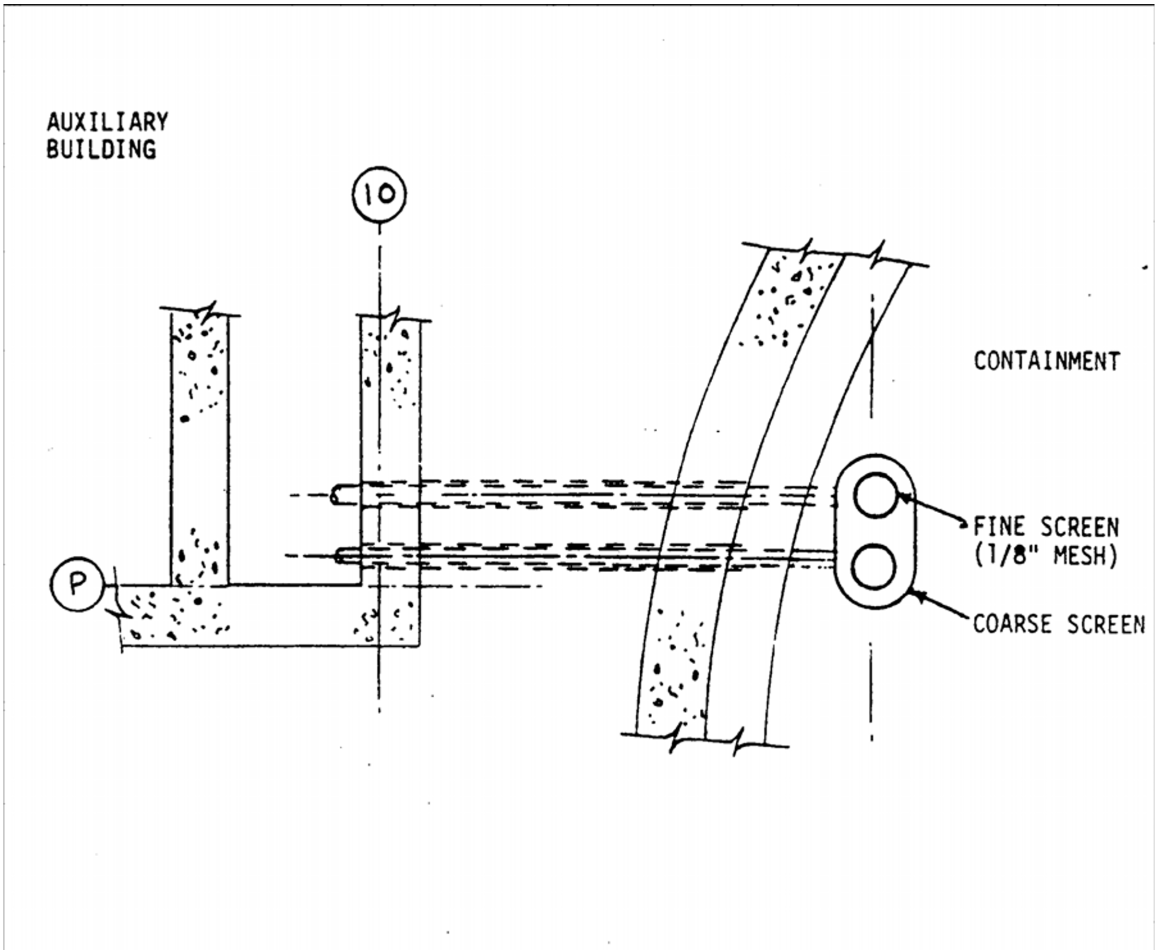
ATTACHMENT 3

SIS DRAINS - ELEVATION



ATTACHMENT 4

SIS DRAINS – PLAN



CONTAINMENT DRAINS - PLAN

ATTACHMENT 5