June 1, 2005

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U.S. Nuclear Regulatory Commission
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Point Beach Nuclear Plant Unit 2
Docket 50-301
License No. DPR 27

Response to Request for Additional Information Regarding
Request for Exigent Review of Heavy Load Analysis

References: 1. NMC Letter to NRC Dated April 29, 2005
2. NMC Letter to NRC Dated May 13, 2005
3. NMC Letter to NRC Dated May 19, 2005

In Reference 1, Nuclear Management Company, LLC (NMC), requested review and approval, in accordance with the provisions of 10 CFR 50.90 and 50.91(a)(6), of proposed amendment to the licenses for Point Beach Nuclear Plant (PBNP), Units 1 and 2, to support a change to the PBNP Final Safety Analysis Report (FSAR) regarding control of heavy loads. The review for PBNP Unit 2 was requested on an exigent basis.

References 2 and 3 submitted supplements to the proposed amendment to provide the results of additional assessments and to incorporate additional technical justification for the proposed amendments. Additionally, Reference (2) retracted the proposed amendment for PBNP Unit 1 and proposed to apply the reactor vessel head (RVH) lift assessment on a one-time basis for the upcoming lift of the Unit 2 RVH.

During telephone conferences between Nuclear Regulatory Commission (NRC) staff and NMC representatives on May 24 and 27, 2005, the staff requested additional information regarding the proposed amendment. The NMC response is provided in Enclosure 1. Enclosures 2 and 3 provide supporting information.

This letter contains no new commitments or changes to existing commitments.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Wisconsin Official.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 1, 2005.

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

Enclosures (4)

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

ENCLOSURE 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING REQUEST FOR REVIEW OF HEAVY LOAD ANALYSIS

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI) regarding Nuclear Management Company (NMC) letters dated May 13 and 19, 2005, which proposed an amendment to the license for Point Beach Nuclear Plant (PBNP) Unit 2, to support a change to the PBNP Unit 2 licensing basis regarding control of heavy loads. The NRC staff's questions are restated below with the NMC response following.

NRC Question 1:

Provide an evaluation establishing that splashing of water injected through the proposed temporary connections to the upper head will not result in loss of water. Stated differently, all water entering via the upper head must be shown to flow into the downcomer or the upper plenum below the elevation of the reactor vessel flange.

NMC Response:

Enclosure 2 provides clarifying diagrams of the expected flowpath of water from the connections of Temporary Modification (TM 2005-008) to the reactor core. The annotations of A, B, C and D refer to the diagrams in Enclosure 2.

Once flow is initiated, water will enter through the top of the head near the center of the dome. Gravity will direct this flow downward towards the upper internals top plate. The upper internals top plate is a flat plate with 33 control rod guide tube top hat sections extending above the top of the plate. Splashing as a result of the relatively low entrance velocity would be captured under the dome of the head and flow down along the inside surface to the periphery interface between the head and the upper internals top plate. The control rod guide tube top hat sections would also serve to dissipate entrance velocity and splashing before the water gets to the periphery. The mating surface between the head and the top of the upper internals top plate prevents water from escaping the head region.

Four carbon steel reactor coolant nozzles are welded to the vessel walls on a centerline below the top surface of the vessel flange. The two 27.5 inch internal diameter (ID) inlet (cold leg) nozzles are located 180° apart. Likewise, the two 29 inch ID outlet (hot leg) nozzles are located 180° apart. Two 3.6 inch ID safety injection nozzles are also welded to the vessel, adjacent to the cold leg nozzles. The reactor vessel has a 132 inch internal diameter.

The holes in the upper internals (A) and the head cooling spray nozzles (B) in the core barrel communicate directly with the down comer region of the core barrel (C). These holes are equally spaced about the periphery of the upper internals and core barrel. The core barrel and hot legs are fit to prevent communication from the hot leg to the downcomer region. During use of TM 2005-008, there is no water pressure to cause flow upwards into the hot leg and the hot leg is tapered towards the vessel. The taper, with the aid of gravity, will direct any leak back into the upper plenum. Therefore, the hot leg is not a path for injected water to escape. The cold leg nozzles represent approximately 13% of the perimeter and the holes (A) are equally spaced about the vessel. It is not expected that splashing would result in water loss out of this 13% of the periphery. Gravity will direct the water into the down comer at a zero degree angle from vertical in a steady stream. There are no internal structures in this region of the core barrel that would deflect water out of the cold leg. The vessel is a minimum of 9 inches thick at the cold legs, which would require the stream at an angle of deflection greater than 12° to reach the outside of the vessel. The cold leg nozzles are tapered toward the vessel, which, with the aid of gravity, would return any misdirected water back toward the downcomer. It is therefore expected that no water would splash out of the cold leg nozzle.

Water entering via the upper head will flow into the downcomer through 24 nozzle openings in the upper internals top plate. These openings are slightly below the top surface of the upper internal top plate. The mating surface between the head and the top of the upper internals top plate keeps the water from escaping the head region. The core barrel and hot legs are fit such that leakage between the downcomer and the nozzle will be minimal. The internal taper (toward the vessel) of the hot leg nozzle will return any leakage back to the upper plenum via gravity.

<u>NRC Question 2</u>:

Show that there is a lip of sufficient height on the top plate structure to trap injected water and reasonably ensure the water will flow into the downcomer or into the upper plenum below the elevation of the reactor vessel flange. If such a lip does not exist, then establish that a configuration exists that reasonably ensures water will flow into the downcomer or into the upper plenum below the elevation of the reactor vessel flange. A cut-away view of the hardware is necessary that clearly shows the configuration and the passages where water will flow downward.

<u>NMC Response</u>:

Enclosure 2 provides clarifying diagrams of the expected flowpath of water from the temporary connections to the reactor core. The annotations of A, B, C and D refer to the diagrams in Enclosure 2.

The head drop analysis considers a concentric drop that results in the head resting on the upper internals top plate. The head is expected to be directed to this position by the guide studs. The core barrel sits 6.75 inches under the vessel flange. The head will rest on the upper internals top plate. The upper internals are held up by the fuel hold down springs. There will be metal-to-metal contact around the entire periphery of the head to upper internals interface. There will be negligible water flow escaping from this mating surface as the pressure head above this interface is small and the less restrictive flow path is through the head cooling spray nozzles. Thus, the flow will be directed into the 24 upper internals peripheral holes (A). If the spring force of the fuel assemblies is insufficient to hold up the head, the upper internals top plate will sit on the core barrel circumferential Bellville spring and all water will be directed down the head cooling spray flow nozzles. The inlets of the spray nozzles are slightly below the top surface of the upper internal top plate, which is 0.813 inches below the reactor vessel flange. If the upper internals top plate rests above the core barrel circumferential Bellville spring due to the fuel assembly spring force, then the water can directly enter the core barrel and have direct communication with the top of the fuel assemblies (D). There will be minimal pressure drop in the flow path from A to D. The flow path from A to C results in a driving-head of approximately 1.5 inches of water to achieve a calculated total flow for all spray nozzles of 12.5 gpm.

<u>NRC Question 3</u>:

Address the result of a slightly asymmetric head drop (i.e., the head comes to rest in a position other than fully aligned with the vessel), assess potential top plate deformation that results, if any, and establish that all water injected through the proposed temporary connections reaches the downcomer or the upper plenum below the elevation of the reactor vessel flange should a slightly asymmetric head drop occur. (Reference Items 1 and 2, above.)

<u>NMC Response</u>:

The NRC staff's request for additional information (RAI) regarding the NMC letter dated May 13, 2005, contained the following statement:

> The staff believes that the 1982 analysis referred to in Wisconsin Electric Power Company letter dated November 22, 1982, establishes an acceptable bounding scenario for evaluation of this event for replacement of the Point Beach, Unit 2 reactor vessel head during the Spring 2005 refueling outage.

The "acceptable bounding scenario" referred to was predicated on the limiting reactor head drop being a concentric (or "symmetric") drop of the reactor vessel head onto the vessel. Such a drop would result in the greatest impact forces to the reactor vessel and supports, and therefore pose the greatest challenge to continued reactor coolant system integrity. In all postulated scenarios of slightly asymmetric drops, it is considered that efforts taken to expand the scope of evaluation beyond a symmetric drop would ultimately be bounded by the previously acceptable bounding scenario noted as having already been reviewed by NRC.

To obtain the upper bounding condition, it must be assumed that the head lines up with the guide studs (approximately 15 feet tall). Since the function of the guide studs is to assure alignment occurs, the head will be appropriately aligned with the vessel and internals if lined up with or on the guide studs. Therefore, the responses to questions 1 and 2 remain valid and bounding.

NRC Question 4:

Provide a detailed assessment of water behavior in the reactor vessel and timing during heatup and boiling including the effect of level swell as a result of thermal expansion during heatup and void development during boiling.

NMC Response:

The following assessment concerns the transient nature of the reactor coolant system (RCS) during the postulated reactor head drop. In this particular scenario, the RCS legs have ceased to be a source of cooling water for the core (but the nozzles remain a vent path). Therefore, the inventory at the start of the incident is the total heat sink until an auxiliary source of cooling water is instituted.

Calculation 2005-0023 (provided in Reference 3) calculated the time to boil and time to core uncovery. The only heat sink credited was the initial RCS inventory below the nozzles in the reactor vessel, excluding the downcomer region and the lower head. All other heat sinks, such as structural steel and fuel materials, were conservatively neglected.

The dynamic nature of the water was not discussed. In reality, as the water in the core absorbs energy from the fuel, its temperature will rise. This temperature rise will cause a corresponding decrease in density. The following descriptive sequence of events is provided to understand the fluid characteristics projected:

Sequence of Events

- Initially, the bulk water temperature is at 100°F.
- The outer cladding surface temperature is also at approximately 100°F due to the high rate of forced convection.
- Once the event occurs, the coolant inventory above the bottom of the nozzles is lost, and the more efficient forced convection stops.
- The heat transfer mechanism changes to free convection.
- The bulk fluid temperature gradually increases (resulting in a decrease in fluid density).
- Eventually, subcooled boiling (efficient heat transfer) along the cladding wall occurs. The voids are quickly collapsed as they enter the bulk fluid, transferring energy to the fluid in the process.
- When the bulk fluid temperature approaches saturation, the voids are sustained and travel upwards to the vapor/liquid interface, increasing in size as they move upwards due to reduced surface pressure on the bubble surface.
- Loss of coolant due to this volumetric expansion primarily consists of a two-phase liquid with a high void content (i.e., a froth), and consequently a small actual loss of

mass. Any loss of mass due to the voids is inconsequential because it (the void) has taken the maximum amount of energy with it.

Manometer Effect

The reactor vessel in this configuration works like a manometer with the downcomer as one leg and the core volume as the other. Both legs are vented to atmosphere through the nozzles. At the beginning of the heatup, both legs have the same external pressure and the same density so their levels are equal. As the water in the core heats up, the density of the water in the core decreases which in turn, causes a rise in level in the core side. However, because the open hot leg nozzles create an upper limit to the level in the core side, some water will spill out through the hot leg nozzles. This will cause a decrease in level on the downcomer side because it remains at a cooler and at a higher density.

The decrease in density from 100°F to 212°F corresponds to an approximate 3.5% increase in volume. Calculation 2005-0023 demonstrated that there was approximately 721 ft^3 of water as a starting inventory. This corresponds to a loss of approximately 25.4 ft^3 (190 gal) of water spillage due to heatup. Some of this water would be replaced by the higher density water of the downcomer until the pressures on both sides equalize. Therefore, Calculation 2005-0023 is conservative in its computation of the time to boil; i.e., some of the water from the downcomer (which is not credited in Calculation 2005-0023) will actually become a portion of the vessel's water inventory and provide additional cooling.

The same phenomenon occurs during the boiling phase. As voids begin to form, the density on the core side of the "manometer" decreases, drawing in more cold (dense) water from the downcomer, thereby increasing the water inventory available for decay heat removal and to maintain the core covered.

This phenomenon occurs because both sides of the reactor vessel are vented. As soon as a void is formed, it escapes and removes its associated quantity of decay heat.

Continuity of Flowpath

Finally, the communication flow path is through the lower core plate diffusion holes This is a good communication flow path as evidenced by the relatively small pressure drop (i.e., large flow areas) through the entire reactor vessel during the much higher flowrates of normal operation. Therefore, flow from the downcomer will easily pass through the lower support plate and core plates.

Conclusion

Calculation 2005-0023 is conservative in both its time to boil and its core uncovery timing.

NRC Question 5:

Provide an evaluation of the effect of the potential head drop on the head assembly upgrade package specifically addressing the ability of the proposed temporary modification to properly function after the head drop.

NMC Response:

The Head Assembly Upgrade Package (HAUP) is connected to the RVH head via three lift legs. An analytical approach was used to determine the effects on these legs, and it was determined that the impact loading on the legs would challenge their structural limits. This potential was further evaluated to understand the effects on TM 2005-008 and its ability to perform its intended function.

The impact of the head on the vessel flange has the potential to result in failures such that the clevis pin connections at the lower end of the three legs would shear, and the legs would begin to buckle. In the case of such a failure, the collapse of the HAUP may be concentric, eventually coming to rest on the head and intervening structures, or the HAUP may be deflected off to the side by an asymmetric collapse of the legs and intervening structures.

If a HAUP collapse occurred, it is expected to progress as follows (a drawing of the HAUP is provided in Enclosure 3).

The impulse loading of the initial impact would be sufficient to shear the lower pin connections. This initial failure may also be sustained long enough to initiate plastic deformation (buckling) of the lifting legs. Some energy would be absorbed in the shearing of the 140 ksi ultimate tensile strength (UTS) steel in the six (6) three inch diameter shear planes of the connections.

After shearing of the pin connections (within 1 inch), a secondary impact would occur when the three lift leg clevis connections bottom against the head lifting lugs. The tops of the lifting lugs are flat (level), and the impact would not impart a bending moment on the legs. If the impact energy were great enough (i.e., from a high enough elevation), the three 6 inch x 8 inch x 0.5 inch rectangular steel lifting legs could then progressively fail by classic Euler buckling. These failures would also absorb some of the energy of the impact through plastic deformation. Any asymmetric collapse of the legs (considered likely since the suspended HAUP is not uniformly balanced about the central axis) would initiate an eccentric collapse.

The HAUP supports the control rod drive mechanism (CRDM) missile shield approximately 32 inches above the top of the 33 CRDM housings (3.8 inch OD, 2.5 inch ID). The next energy absorbing impact would be as the missile shield contacts the tops of these housings. The upper housings are relatively long, thin columns that can also be expected to collapse via Euler buckling and gross deformation. The upper ends of these housings are tied together by the Rod Position Instrumentation (RPI) plate. This connection tends to prevent independent buckling of the individual housings (i.e., constrains them against independent translation). The lower ends of the upper CRDM housings are constrained against rotation and translation by the closely spaced CRDM magnet coil stacks, and the CRDM head penetrations themselves. This creates an assembly that is considerably more resistant to buckling than the sum of the individual housings alone, and one that would tend to amplify an asymmetric collapse to more effectively laterally deflect the descending HAUP mass. Additionally, the energy required to fully collapse all of the columns is greatly increased by the effective fixation of the lower ends.

The next impact / energy absorbing section of intervening structure is a combination of the CRDMs themselves and the shield ring. These two sets of structures are quite substantial. Their tops are approximately 165 inches below the top of the CRDM upper housing.

At this elevation the CRDM housings are flared out to a larger diameter (6.5 inch OD, 0.62 inch thick wall) for a relatively short distance (approximately 45 inches). The CRDM magnets are around this section of housing to create a close-fitting grid of square blocks. This relatively solid structure is not susceptible to buckling, and would tend to transmit any axial forces straight downward to the head, absorbing any overload by localized plastic deformation.

The radiation shield ring is a vertical cylinder fabricated from 1.25 inch bent stainless steel plate. The ring is made up of two layers of segments bolted together with splice plates at the vertical seams. It is punctuated by various access doors for inspection of the upper head penetrations and to make up instrumentation and reactor vessel level indication system (RVLIS) connections. This heavy steel ring assembly rests directly on the head lifting lugs, and represents an extremely stiff and robust stop for any collapse of the HAUP onto the vessel head.

The two head penetrations proposed for TM 2005-008 injection points are both located below the CRDM magnet stacks and the top of the lower shield ring. The hose connection to the head vent piping is made where the head vent piping terminates outside the lower shield ring. As such, only an asymmetric collapse of the HAUP into the quadrant containing the TM 2005-008 head vent connection is likely to cause damage to this connection.

Conversely, the TM 2005-008 connection to the RVLIS piping extends upward through the shield rings, with the hose emerging from the ventilation shroud above the rings. Only a concentric collapse of the HAUP, or an asymmetric collapse into the quadrant occupied by the hose as it exists the shroud would be likely to damage this connection.

Since the head vent connection and routing is not in the same quadrant as the RVLIS connection and routing, and utilizing engineering judgment to assess the probability of loss of both temporary modification lines in this postulated scenario, it is considered highly unlikely that any collapse of the HAUP would render both temporary hose connections inoperable.

NRC Question 6:

Provide a docketed copy of the bottom mounted instrument tube analysis previously provided informally to the NRC staff.

NMC Response:

The bottom mounted instrument tube analysis, previously provided informally to the NRC staff, is provided in Enclosure 4. A brief description of that analysis follows.

The analyses were performed by inputting a vertical nozzle movement at node 5 and increasing it until maximum stresses in the conduit approached 3.0 S_c, which equates to 56,400 psi based on the line material (A213 TP304).

It was postulated that the bottom of the bend below the reactor pressure vessel (RPV) would contact the floor first, after being allowed to displace 1 inch before being restrained. This contact was modeled as a vertical support at node '30 F', with a gap below the conduit of 1 inch and a large gap above the conduit to allow unrestricted upward movement. The analysis was then performed again, noting the vertical displacements of the conduit on the horizontal run. Any location exceeding 1 inch downward was then restrained in subsequent runs. The final configuration shown in the preceding plot has supports at nodes '30 F' and '85 N' simulating contact with the floor at those two locations. All the other points either deflect upward, or deflect less than 1 inch downward.

It is postulated that all 36 lines will move in phase, reducing contact between individual lines, when the RPV displaces downward.

The AutoPIPE output is found in Attachment A of the calculation provided within Enclosure 4. The maximum stress for a 23 inch downward displacement of the RPV is 55,385 psi at node '30 F'. This is below the 56,400 psi allowable of 3.0 S_c.

NRC Question 7:

Provide the technical justification for assuming free movement of the bottom mounted instrument (BMI) tubes given the very limited clearance between the tubes and the containment floor.

NMC Response:

The calculation utilizes an 'AUTOPIPE' pipe model to establish the maximum vessel displacement required to over stress the BMI tubes. The model uses a thermal growth methodology, taking into account the gaps at the supports. This calculation was developed to support the conclusion reached in the May 13, 2005, letter, which states that based on engineering judgment, the BMI connections remain intact. The clearance between the bottom of the first support, which is a spring can support, and the floor is 1 inch. The analysis conservatively assumes that the entire run is only allowed to deflect 1 inch downward before floor contact is made.

<u>NRC Question 8</u>:

Provide the technical justification for ignoring the likelihood of bottom mounted instrument tube restraint due to friction, recognizing the potential for such restraint to result in tube crimping and possible cracking.

<u>NMC Response</u>:

The supports for the BMI instrumentation are U-Bolt supports, which have design gaps of between 3/16 inch to 1/4 inch in the vertical direction, and 1/8 inch to 3/16 inch in the horizontal direction. Therefore, the U-Bolts would permit axial movement. However, as requested by the staff, an informal computer analysis was performed that assumed restraining the BMI tubes axially at the first spring restraint. The downward deflection of the reactor vessel for this condition was approximately 6 1/2 inches of travel to maintain stresses below the 3.0 S_c limit. With 1 inch of axial movement of the BMI allowed, the downward deflection was 7 ¾ inches of travel to maintain stresses below the 3.0 S_c limit. This provides an upper bound for the effects of friction at the supports.

<u>NRC Question 9</u>:

Provide the technical justification for ignoring the likelihood of high stresses in the bottom mounted instrument tubes resulting in weld cracking and subsequent leakage at the tube to reactor vessel junction.

<u>NMC Response</u>:

The calculation associated with the BMI analysis (Enclosure 4) uses a stress intensification factor (SIF) of 1.3 to account for concerns with stresses on welds, at the connection to the reactor vessel, and at the coupling welds located along the BMI tubes. This SIF is in accordance with ANSI B31.1-1967 to account for the higher stresses at fillet welded joints.

<u>NRC Question 10</u>:

The analysis of the bounding radiological consequences of a head drop includes the postulated radiological release from emergency core cooling system (ECCS). The doses due to the head drop are scaled to and compared to the current licensing basis analysis of the large-break loss-of-coolant accident (LOCA), as taken from Section 14.3.5 of the Point Beach Final Safety Analysis Report (FSAR). Table 14.3.5-5 gives different ECCS leakage rates for calculation of offsite doses (800 cc/min) vice the calculation of control room doses (400 cc/min). In a conference call on May 24, 2005, Point Beach staff clarified that 400 cc/min is the ECCS leakage administrative limit.

Although there is no specific guidance on the dose analysis of a head drop, some guidance on the ECCS leakage pathway in LOCA analyses can be considered useful. Section 4.2 in Appendix A to Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (ML031490640)", gives guidance on the assumption for ECCS system leakage, and states that the ECCS systems leakage factor of two multiplier is used to account for increased leakage in the systems over the duration of the accident and between surveillances or leakage checks. Provide a technical basis for why this multiplier has not been used as an assumption for the calculation of both the offsite and control room doses due to the head drop.

<u>NMC Response</u>:

The dose evaluation provided in NMC letter dated May 19, 2005, considered the guidance of Regulatory Issue Summary (RIS) 2001-0019 and Position 1.3.2, "Reanalysis Guidance", of Regulatory Guide (RG) 1.195, as well as other sections of this RG. RIS 2001-0019, Position 1, states that an assumption made in a licensee's analysis supporting a docketed amendment request to be part of the current design basis if the staff relied upon that assumption when evaluating whether the NRC requirements were met in granting the licensing amendment. Therefore, since the RVH drop for PBNP is an accident initiator of a LOCA, the CLB LOCA radiological accident analysis was reviewed to determine the portions of the analysis that would be impacted by this event. The CLB LOCA radiological accident analysis for the control room does not include the factor of two multiplier on ECCS. The CLB LOCA radiological accident analysis was incorporated into the FSAR under NRC safety evaluation (SE) dated July 9, 1997. All other design basis radiological accident analyses were approved under NRC SE dated July 1, 1997.

In addition, Position 1.3.2 of RG 1.195 provides the NRC staff expectation that a licensee evaluate all impacts of a proposed change on the facility's design and licensing bases and to update the affected analyses and the design bases appropriately. The RVH drop event does not physically impact the structure or components of the ECCS, it does not require that the ECCS be operated in a manner that is outside of the current design and licensing basis, and it does not require a change to the current licensing basis methodology for assessing a radiological release due to ECCS leakage. Therefore, it was concluded that no change to the licensing basis assumption for ECCS leakage needed to be taken into consideration because the proposed licensing change does not impact the analysis such that the results or the conclusions drawn are no longer valid. In other words, the LOCA analysis is not considered to be affected as defined in RG 1.195 Position 1.3.2.

Moreover, the current licensing basis LOCA ECCS leakage is based on the administrative limit implemented via the Leakage Reduction and Preventative

Maintenance Program (LRPM). This limit was established via License Amendments 174/178 (dated July 9, 1997). FSAR Section 6.2 discusses this program. Leakage measurements for most of the components of the system that are part of the LRPM for Unit 2 have been obtained during the current Unit 2 outage. At present, the leakage value for the current conditions is 161 cc/min, which is greater than a factor of two lower than the administrative limit. The current Unit 2 ECCS leakage value includes past measurement leak rates for those components not yet tested this outage. Based on a review of leakage histories for these components, the final measured leakage value for Unit 2 is expected to remain less than 200 cc/min.

The intent of the dose assessment evaluation provided was to provide reasonable assurance via a scoping analysis that for the postulated Unit 2 reactor vessel head drop over the Cycle 28 reload core at 30-days post-shutdown, the dose consequences off-site and to the control room are clearly bounded by the current licensing basis LOCA radiological accident analysis and do not create a more severe hazard. This evaluation demonstrates that the LOCA remains the limiting event for control room habitability, as well as, for offsite locations. The evaluation provided is not intended to support a revision to the results of a current radiological accident analysis nor is it intended to support incorporation of a new design basis accident. If the administrative limit for ECCS leakage had been doubled, the dose to the control room post-RVH drop would also double. However, the RVH drop results would still be bounded by the current licensing basis LOCA since doubling RVH drop ECCS leakage dose remains significantly less than the LOCA due to the fact that the source term for the RVH drop event is about a factor of 75 less than the CLB LOCA.

<u>NRC Question 11</u>:

The NRC staff must make a finding as to whether the licensee has shown through control room dose analyses that General Design Criteria (GDC)-19 has been met for the proposed license amendment. The doses due to the head drop are scaled to and compared to the current licensing basis analysis of the large-break LOCA, as taken from Section 14.3.5 of the Point Beach FSAR. The FSAR LOCA dose analysis does not take into account the results of control room envelope unfiltered inleakage testing, nor does the scaling calculation. In response to the information requests in GL 2003-01, by letter dated September 29, 2004, the licensee committed to supply the final control room envelope testing inleakage results to the NRC as required to support any licensing actions. Provide a technical basis for not using the control room envelope unfiltered inleakage testing results in the analysis of the offsite and control room doses due to the head drop.

<u>NMC Response</u>:

The commitment contained in the NMC letter dated September 29, 2004, was made in regards to the NRC expectation in GL 2003-01 that licensees, who are unable to confirm that the most limiting unfiltered inleakage into the control room envelope (CRE) is no more than the value assumed in the design basis radiological analysis for control room habitability, develop and implement corrective actions as required by 10 CFR 50, Appendix B. Since the LOCA is the limiting design basis radiological analysis for control room habitability, final resolution of GL 2003-01 for PBNP will require reconstitution of the LOCA radiological design basis analysis. It is recognized that in support of this final resolution, demonstration of the most limiting accident for control room habitability (CRH) may require a re-evaluation of all design basis accidents.

The dose evaluation provided in the NMC letter dated May 19, 2005, demonstrates through comparison of the RVH drop source term to the CLB LOCA ECCS leakage source term that the RVH drop event does not result in a more severe hazard to the control room than the LOCA; therefore, the LOCA remains the limiting event for CRH. Similar to the logic provided in response to Question 10, the RVH drop does not impact or create a cause to change the unfiltered inleakage assumed in the LOCA accident analysis. The numerical dose values presented in the evaluation were to provide a means to demonstrate a margin of safety which demonstrates reasonable assurance that the event does not result in a significant increase in the consequences of any accident previously evaluated. Incorporation of the measured unfiltered inleakage does not change this conclusion since it is based on a scaled LOCA dose, which is the design basis radiological analysis for control room habitability for PBNP.

This conclusion can be further clarified by a qualitative review of the constituents of the control room dose via the ECCS leakage pathway. The control room operator inhalation and whole body dose (due to activity internal to the control room) is driven by the amount of activity that passes through the control room ventilation filter into the control room. When the control room emergency filtration system (CREFS) is in operation (mode 4 of the control room ventilation system), it is assumed that 4950 cfm of outside air is supplied to the control room to provide filtered air which will pressurize the control room. Since the efficiency of the CREFS filters is 95% for elemental/organic and 99% for particulate, the activity is entering the control room via the ventilation system at an estimated rate of 250 cfm for elemental/organic and 50 cfm for particulate. The current licensing basis CRH analysis assumes that while in mode 4, the unfiltered inleakage is 10 cfm. The CLB ECCS leakage pathway is assumed to contain only elemental iodine. Therefore, under the current licensing basis analysis, the dose to the operator via the ECCS leakage path is primarily due to the elemental iodine activity delivered through the ventilation system.

Recent tests of unfiltered inleakage to the control room while in mode 4 determined that unfiltered inleakage is approximately 100 cfm. This information was previously discussed in NMC letters to the NRC dated December 5, 2003 and September 29, 2004. If the measured unfiltered inleakage value were to be incorporated into the dose analysis, dose due to elemental iodine would increase proportionally to the ratio of x/y, where x is the rate of activity delivered to the control room including measured unfiltered inleakage (250 cfm + 100 cfm) and y is the rate of activity delivered under current licensing basis assumptions for unfiltered inleakage (250 cfm + 10 cfm). Therefore, the elemental dose would only increase by a factor of 1.35 (350 cfm / 260 cfm). Since the ECCS leakage pathway is assumed to contain only elemental activity; the doses would, at most, increase by a factor of 1.35.

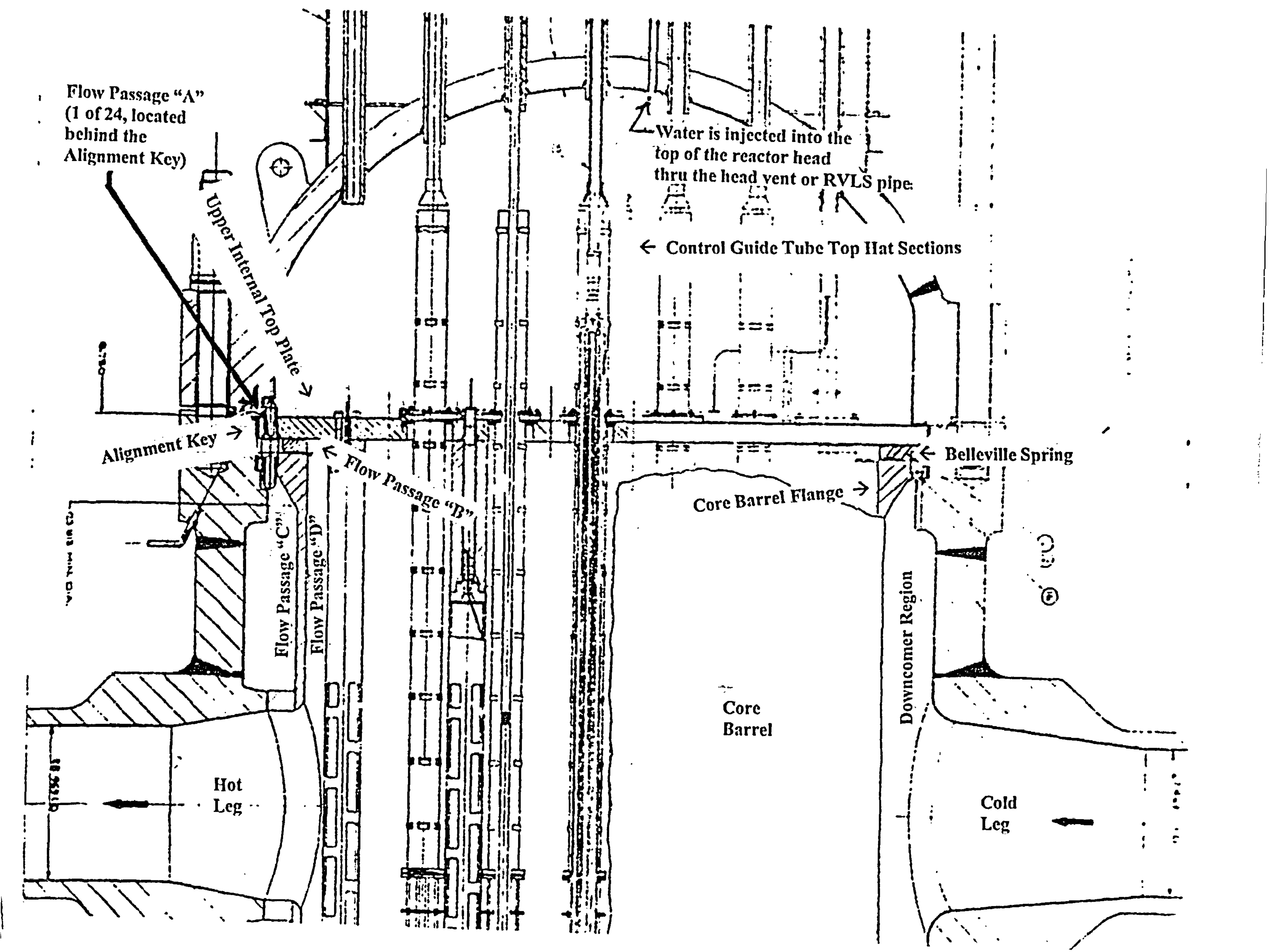
Therefore, if the measured unfiltered inleakage had been included (factor of 1.35 increase) as well as doubling the administrative limit for ECCS leakage value (factor of two increase), the dose to the control room operator post RVH drop would increase by only a factor of 2.7 (i.e., 2 x 1.35) and the conclusion that the RVH drop is bounded by the LOCA remains valid. As noted above, the ECCS leakage for the current conditions is expected to be greater than a factor of two lower than the administrative limit. Therefore, based on current conditions the dose to the operator would only increase by a factor of 1.35.

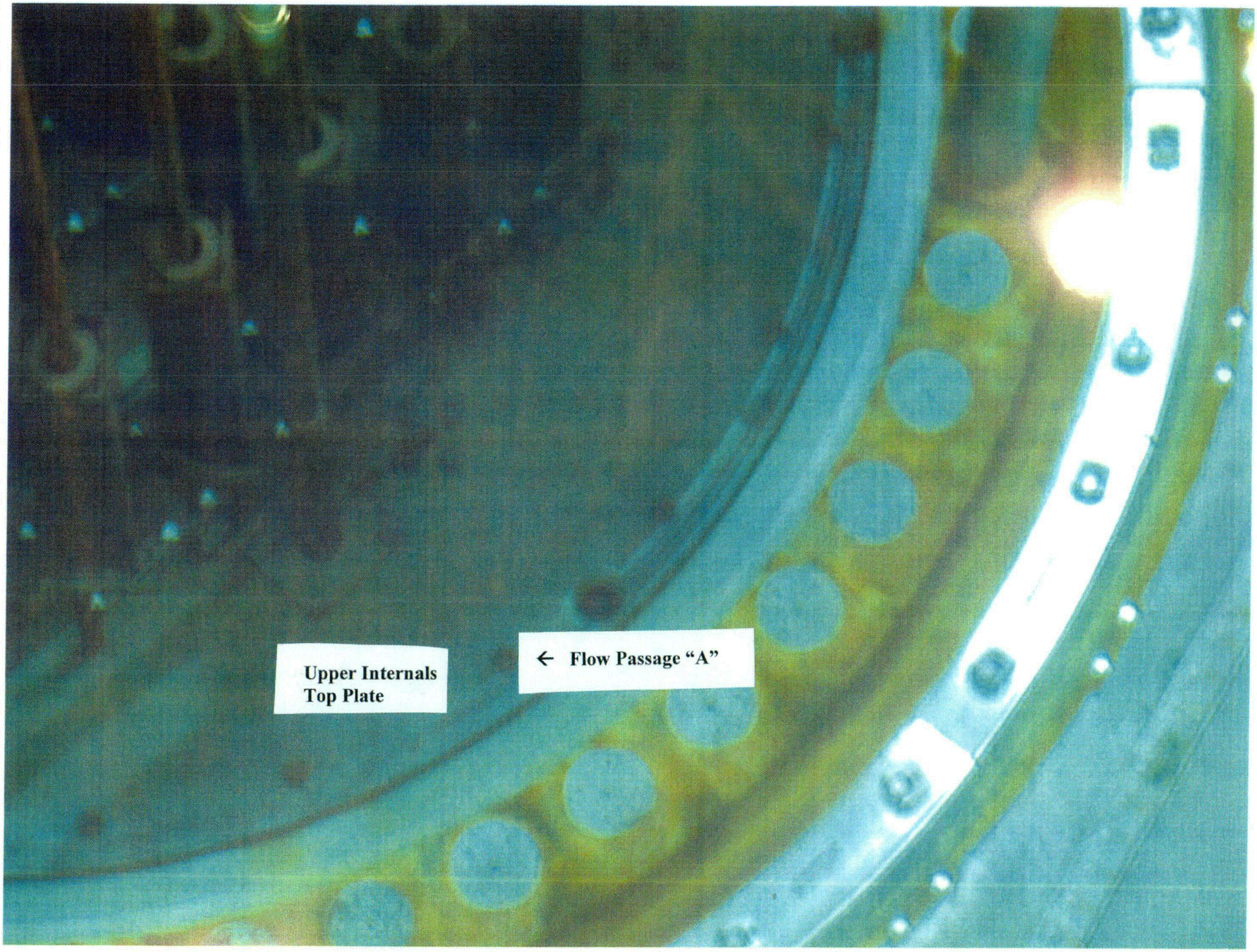
The source term assumed for the Unit 2 Cycle 28 RVH drop event provides a significant margin of safety such that incorporating these other effects that are not driven by the event itself do not change the conclusion that the LOCA radiological dose consequence analysis remains bounding.

ENCLOSURE 2

DIAGRAMS OF WATER FLOWPATHS
FROM THE TEMPORARY CONNECTIONS TO THE REACTOR CORE

(2 pages follow)


Flow Passage "A" (1 of 24, located behind the Alignment Key)
Water is injected into the top of the reactor head thru the head vent or RVLS pipe.
Upper Internal Top Plate
Control Guide Tube Top Hat Sections
Alignment Key
Flow Passage "B"
Belleville Spring
Core Barrel Flange
Flow Passage "C"
Flow Passage "D"
Downcomer Region
Core Barrel
Hot Leg
Cold Leg



Upper Internals
Top Plate
← Flow Passage "A"

ENCLOSURE 3

DIAGRAM OF THE HEAD ASSEMBLY UPGRADE PACKAGE

(1 page follows)

Point Beach RRVCH with Head Assembly Upgrade Package