June 1, 2010

NRC 2010-0075
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets No. 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 265
Revision to the Reactor Vessel Head Drop Methodology

References:
(1) NRC letter to Nuclear Energy Institute, dated September 5, 2008, Safety Evaluation Regarding NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0 (ML082410532)
(2) NRC letter to Nuclear Management Company, LLC, dated September 23, 2005, Point Beach Nuclear Plant Units 1 and 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis into the Final Safety Analysis Report (TAC Nos. MC7650 and MC7651) (ML052560089)
(3) NRC letter to Nuclear Management Company, LLC, dated January 12, 2006, Point Beach Nuclear Plant, Units 1 and 2 - Revision to Safety Evaluation for Amendment Nos. 220 and 226 (TAC Nos. MC7650 and MC7651) (ML052850005)

Pursuant to 10 CFR 50.90, NextEra Energy Point Beach, LLC (NextEra), hereby requests an amendment to Renewed Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP), Units 1 and 2. The proposed amendment consists of revising the current license basis regarding a postulated reactor vessel head (RVH) drop event to conform to the NRC-endorsed guidance of Nuclear Energy Institute (NEI) 08-05 (Reference 1).

The proposed change to the license basis will revise Final Safety Analysis Report (FSAR) Chapter 14.3.6, Reactor Vessel Head Drop Event. The current license basis for the RVH drop methodology was approved by the Commission via issuance of Amendment Nos. 220 and 226 (Reference 2), and subsequent revision (Reference 3).

A dynamic analysis, conforming to NEI 08-05 guidance, demonstrates that a postulated RVH drop event would not result in a breach of the reactor coolant system (RCS). Therefore, core cooling would not be compromised and a coolable geometry would continue to be maintained in the core. Revising the current license basis to conform to the NRC-endorsed guidance of NEI 08-05 represents a change in the method of evaluation for a postulated RVH drop event at PBNP. The proposed change, therefore, requires Commission approval.

Enclosure 1 contains the description and analysis of the proposed change. Attachment 1 of Enclosure 1 provides a mark-up of the proposed changes to FSAR Chapter 14.3.6.

Approval of the proposed amendment is requested by June 1, 2011. NextEra will implement the amendment within 30 days of Commission approval.

NextEra has evaluated the proposed amendment and has determined that it does not involve a significant hazards consideration pursuant to 10 CFR 50.92. The Plant Operations Review Committee has reviewed the proposed license amendment request.

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

This letter contains no new Regulatory Commitments and no revisions to existing Regulatory Commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on this day, June 1, 2010.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosure/Attachment

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

ENCLOSURE 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 265
REVISION TO THE REACTOR VESSEL HEAD DROP METHODOLOGY

DESCRIPTION AND EVALUATION OF CHANGES

ATTACHMENT:

1. Proposed Final Safety Analysis Report Chapter 14.3.6 Changes (Mark-up)

1.0 SUMMARY DESCRIPTION

This license amendment request supports a revision to the Point Beach Nuclear Plant (PBNP) license basis regarding a postulated reactor vessel head (RVH) drop event. The current license basis assumes failure of the reactor coolant system (RCS) boundary. The failure of the RCS boundary would be caused by the predicted maximum downward displacement of the reactor vessel which would sever all 36 bottom-mounted instrument (BMI) conduits (tubes), causing a loss of RCS inventory. The current license basis is documented in Final Safety Analysis Report (FSAR) Chapter 14.3.6, Reactor Vessel Head Drop Event.

The proposed revision to the license basis incorporates an updated analysis that conforms to Nuclear Energy Institute (NEI) 08-05 (Reference 6.1). The new analysis demonstrates that a postulated RVH drop would not result in a loss of RCS inventory caused by an RCS boundary failure, since BMI conduits would remain intact. A dynamic analysis of the BMI conduits has been performed. NextEra determined that in the limiting case, stresses on the BMI conduits were less than those allowed by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections II and Section III Division 1 Appendix F, 1998 Edition through 2000 Addenda.

2.0 DETAILED DESCRIPTION

The postulated RVH drop accident involves the concentric drop of the RVH onto the reactor vessel flange from a height of 26.4 feet. The resultant impact displaces the reactor vessel downward, which creates the potential for damage to RCS piping and tubing directly or indirectly connected to the reactor vessel, thereby creating the potential for a decrease in RCS inventory.

The proposed change to the license basis will revise FSAR Chapter 14.3.6. Analysis of the postulated reactor vessel head drop event was incorporated into the FSAR pursuant to the NRC Safety Evaluation (Reference 6.2) and subsequent revision (Reference 6.3), for License Amendments Nos. 220 and 226 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for PBNP Units 1 and 2, respectively. Proposed changes to FSAR Chapter 14.3.6 are provided as Attachment 1 of this enclosure.

Current FSAR Chapter 14.3.6 information is being removed that relates to the current license basis RVH drop event analysis, which assumed the predicted maximum downward displacement of the reactor vessel would sever all 36 BMI conduits. FSAR Chapter 14.3.6 will be updated to reflect the revised analysis results, which show the maximum downward displacement will not damage RCS piping and a coolable geometry will be maintained in the core. The proposed revision to FSAR Chapter 14.3.6 includes removing the radiological analysis. In accordance with the NRC-endorsed methodology contained in NEI 08-05 (Reference 6.1), which states, "Previous evaluations have indicated that the consequences of impacts between the upper vessel internals and the fuel were not significant with respect to public health and safety," a revised radiological analysis was not performed.

A dynamic analysis, conforming to NEI 08-05 (Reference 6.1), demonstrates that the BMI conduits would remain intact, and thus, RCS integrity would not be compromised. BMI conduits were qualified for the RVH drop event by showing that the calculated maximum primary stress intensities were below the allowable ASME Code limit for Level D conditions (ASME Boiler and Pressure Vessel Code, Sections II and Section III Division 1 Appendix F, 1998 Edition through 2000 Addenda).

3.0 TECHNICAL EVALUATION

The current license basis static analysis of the PBNP postulated RVH drop analysis did not evaluate the dynamic input and response of the BMI conduits. The current license basis analysis assumed the downward force of the dropped RVH, coupled with the limited clearance between the BMI conduits and the floor would sever all 36 BMI conduits. The structural integrity of the BMI conduits is not considered in the current license basis structural integrity analysis.

Pursuant to the NRC-endorsed guidance of NEI 08-05, an evaluation of the BMI conduits for the postulated closure head assembly (e.g., the RVH) drop events was performed using finite element models of BMI conduit numbers 32 and 29, which are identified in Figure 1 below. The finite element model was generated in ANSYS®, Version 11.0. Acceptability is based on maintaining the structural integrity of the BMI conduits such that core cooling will not be compromised.

Only two models were required; one for conduit number 32 and one for conduit number 29. BMI conduit numbers 32 and 29 were analyzed because they are the conduits with the shortest and longest overall lengths, respectively. It is assumed that all of the BMI conduits experience the same time-history transient effects due to the RVH drop. Therefore, selecting the shortest and longest BMI conduits will give a bounding range of the stresses experienced by all of the BMI conduits during the RVH drop.

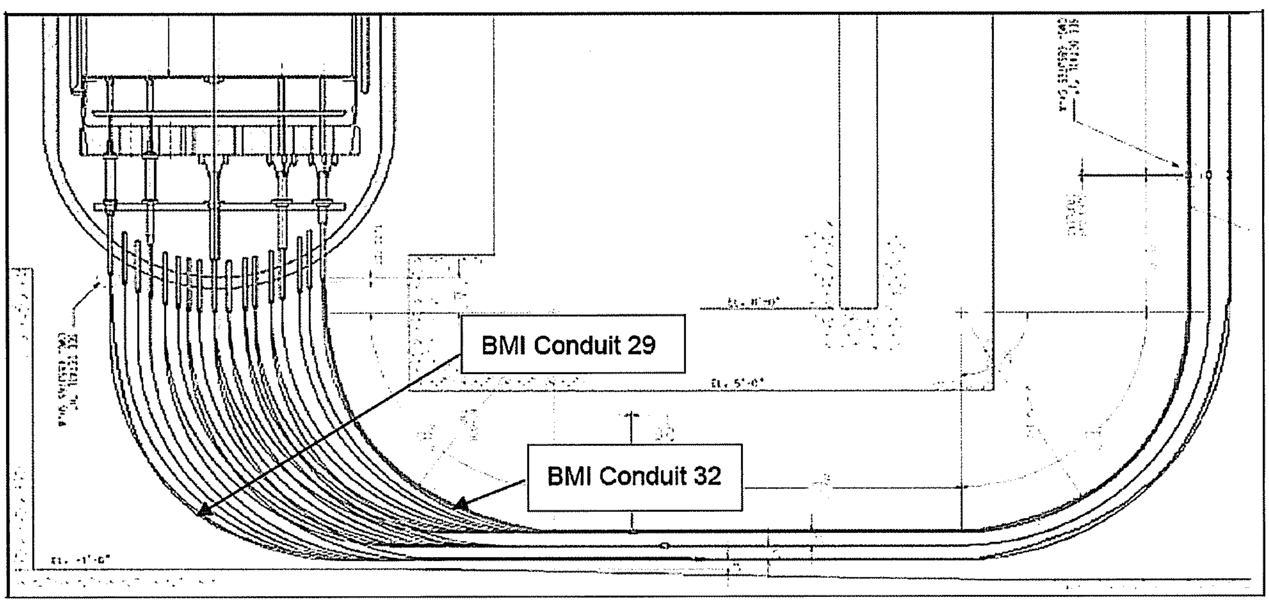

Figure 1

The displacement time-histories calculated in the current license basis RVH drop analysis were applied to the BMI conduit models. The displacement time-histories were originally applied to the BMI conduit models at the reactor vessel-BMI interface; however, this resulted in incorrect applied accelerations. ANSYS® determined accelerations based on the input displacement time-histories. "Noise" in the displacement time-histories caused large, unrealistic accelerations to be applied to the models.

A spring-mass system was added to the BMI conduit finite element models between node 10000000 (node representing the reactor vessel) and node 1 (BMI nozzle to BMI conduit interface) to filter out the high frequency noise. The natural frequency of the spring was selected such that the high frequency noise would be eliminated without impacting the response of the conduit. A natural frequency of 100 Hz was selected for the spring-mass system. This frequency will filter out the high frequency noise without impacting the input frequency of 17.24 Hz and the BMI conduit natural frequency of 17.686 Hz.

The spring stiffness was made sufficiently high to ensure that the BMI conduit would follow the input displacement time-history. A spring stiffness of 100,000,000 lb_f/in was selected. The mass of the system was calculated from the natural frequency and stiffness of the spring-mass system using Equations 1 through 3, below. The computed mass assigned to the spring-mass system was 253.3 lb_f-s^2/in. The mass value is large relative to the BMI conduit mass (0.498 lb_f-s^2/in), minimizing feedback from the BMI conduit into the applied load. Therefore, the output response of the spring-mass system is equivalent to the input displacement time-history.

Equation 1:	$\omega_n = \sqrt{\dfrac{K}{m}}$	Natural Frequency (rad/s)
Equation 2:	$f = \dfrac{1}{2 \cdot \pi} \omega_n$	Frequency (Hz)
Equation 3:	$m = \dfrac{K}{(2 \cdot \pi \cdot f)^2}$	Mass for Spring System (lb_f-s^2/in)

The finite element models were constrained to represent the supports derived from walk-down information. The displacement time-histories for Units 1 and 2 were applied through node 10000000 to the BMI nozzle location at node 1. The displacement time-histories were used to determine the responses of the BMI conduits to the postulated RVH drop events defined in the current license basis RVH drop event. Then, the maximum stress intensity was calculated at each node for the entire dynamic analysis for both models using Equation 4. The maximum value of each model was compared to the appropriate ASME Code allowable stress to determine acceptability.

Equation 4:	$\sigma_{\text{intensity}} = \dfrac{P}{A} + \sqrt{\left(\dfrac{M_x \cdot c}{I}\right)^2 + \left(\dfrac{M_z \cdot c}{I}\right)^2}$	Stress Intensity (psi)

Finally, static and modal analyses were performed to better understand the responses generated by the dynamic BMI conduit finite element models.

Table 1 summarizes the results of the revised analysis, assuming that the BMI conduits do not contact the floor and the large deflection option within ANSYS® is not used. Additional analyses, consistent with the guidance contained in NEI 08-05, modeled three additional cases to demonstrate that the stress that would be experienced by the conduits would remain within the allowable limits. These three cases are:

1. Activating the large deflection option within ANSYS®;
2. Modeling contact between the BMI conduits and the underlying floor; and,
3. A combination of floor contact and having the large deflection option within ANSYS® activated.

Results of the additional three cases are contained in Tables 2, 3 and 4, respectively. Tables 3 and 4 only contain stress intensities for BMI conduit 29, because it was determined that BMI conduit 32 does not deflect enough to contact the floor.

The maximum stresses that the BMI conduits experience were determined to be within the allowable limits, as presented below. Therefore, it is concluded that the BMI conduits are acceptable for the postulated RVH drop event.

Table 1: Maximum Stress Intensity Results for BMI Conduits, without floor contact and without use of the large deflection option

BMI Conduit Number	Stress Category	Location	Time (seconds)	Stress Intensity (psi)	Allowable Stress (psi)	Margin (%)
Unit 1 Conduit 32	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2850	10,410	52,500	80.17
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3422	62,008	67,500	8.14
Unit 2 Conduit 32	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	9,940	52,500	81.07
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3416	61,897	67,500	8.30
Unit 1 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.285	11,430	52,500	78.23
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3502	61,569	67,500	8.79
Unit 2 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.285	10,910	52,500	79.22
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3494	61,288	67,500	9.20

Table 2: Maximum Stress Intensity Results for BMI Conduits, with the large deflection option within ANSYS® activated

BMI Conduit Number	Stress Category	Location	Time (seconds)	Stress Intensity (psi)	Allowable Stress (psi)	Margin (%)
Unit 1 Conduit 32	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	10,190	52,500	80.59
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3381	64,894	67,500	3.86
Unit 2 Conduit 32	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	9,742	52,500	81.44
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3371	64,794	67,500	4.01
Unit 1 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	11,249	52,500	78.57
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3451	64,647	67,500	4.21
Unit 2 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	10,741	52,500	79.54
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3446	64,342	67,500	4.68

Table 3: Maximum Stress Intensity Results for BMI Conduit 29, modeling contact between the BMI conduits and the underlying floor

BMI Conduit Number	Stress Category	Location	Time (seconds)	Stress Intensity (psi)	Allowable Stress (psi)	Margin (%)
Unit 1 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	11,436	52,500	78.22
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3001	62,142	67,500	7.94
Unit 2 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	10,915	52,500	79.21
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3	61,430	67,500	8.99

Table 4: Maximum Stress Intensity Results for BMI Conduit 29, modeling contact between the BMI conduits and the underlying floor with the large deflection option within ANSYS® activated

BMI Conduit Number	Stress Category	Location	Time (seconds)	Stress Intensity (psi)	Allowable Stress (psi)	Margin (%)
Unit 1 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	11,249	52,500	78.57
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.3000	60,367	67,500	10.57
Unit 2 Conduit 29	Membrane Stress	BMI Nozzle to BMI Conduit Interface	1.2846	10,741	52,500	79.54
	Membrane plus Bending Stress	BMI Nozzle to BMI Conduit Interface	1.2998	59,652	67,500	11.63

Conformance with NEI 08-05 (Reference 6.1) represents a change in the method of evaluation for a postulated RVH drop event at PBNP. The dynamic analysis of the potential interaction between the BMI conduits and the underlying concrete floor determined that there was reasonable assurance that the BMI conduits, and thus, the RCS boundary, would remain intact following a postulated drop of the RVH onto the reactor vessel.

NextEra evaluated the proposed changes to the PBNP license basis against the Industry Analysis contained in Table 1, Comparison of Industry Initiative with NUREG-0612 of NEI 08-05 (Reference 6.1). The results of that analysis, which show the revised methodology conforms to the NRC-endorsed methodology contained in NEI 08-05 (Reference 6.1), is provided below.

<u>NEI 08-05, Table 1 Initiative Analysis</u>

Demonstrate that after the reactor vessel head drop, the core remains covered with coolant and sufficient cooling is available.

<u>NextEra Evaluation</u>

If the downward deflection of the reactor vessel is great enough, there is a concern that the BMI conduits, located below the reactor vessel, may be able to impact the concrete floor slab and/or be subjected to significant stress concentrations, particularly where they join the reactor vessel. As such, it was necessary to analyze the stress conditions in these conduits that may be caused by a postulated RVH drop event.

The BMI conduit models used the displacement time history of downward vertical displacement of the reactor vessel from the current RVH drop analysis as an input. The current license basis RVH drop was previously determined to be acceptably modeled to provide appropriate outputs for the postulated event, and was accepted by the NRC. Therefore, use of the displacement output from that calculation was appropriate when considering the effects of a postulated RVH drop on the BMIs.

The BMI conduit model analysis concluded that in the limiting conduit cases, both the direct membrane stresses and the combination of membrane plus bending stress were lower than the ASME Code allowable stresses.

Additional analysis modeled three additional cases to demonstrate that the stress that would be experienced by the conduits would remain within the ASME Code maximum allowable limits:

1. Activating the large deflection option within ANSYS®;
2. Modeling contact between the BMI conduits and the underlying floor; and,
3. A combination of floor contact and having the large deflection option within ANSYS® activated.

NEI 08-05 requires the use of the large deflection option when modeling components within the impact load path. While the conduits are not in the impact load path, adherence with the approved guidance for the load path components provides consistency of methodology. Modeling of potential conduit/floor interaction may result in a more limiting condition.

Inclusion of the large deflection option results in more severe stresses; although within the ASME Code allowable limits. Modeling of floor contact without the large deflection option reduced the calculated stresses to approximately the point that they were without either floor contact or large deflection turned on, and modeling both floor contact and large deflection resulted in the lowest calculated stresses. In no case were the applicable ASME Code allowable stresses exceeded.

Since the maximum acceptable stress limits would not be exceeded in the BMIs, RCS boundary integrity remains intact. Therefore, coolant inventory would be maintained, and the functional cooling capability of connected systems required by Technical Specifications to be OPERABLE and in operation (specifically the residual heat removal system) would not be impaired. Therefore, the core would remain covered and sufficient cooling would be available.

NEI 08-05, Table 1 Initiative Analysis

The reactor vessel head drop is concentric and impacts directly on the vessel flange.

NextEra Evaluation

The current license basis RVH drop accident involves the concentric drop of the RVH onto the reactor vessel flange. The displacement time-histories calculated in the current license basis RVH drop analysis were applied to the BMI conduit models.

NEI 08-05, Table 1 Initiative Analysis

The reactor vessel head is dropped directly above the vessel at the maximum height controlled by plant procedures. In some plant procedures, the reactor vessel head may be moved horizontally and still be over the flange, and then lifted further. The maximum drop height is determined by the maximum height above the flange while the reactor vessel head center of gravity is still within the flange radius of over the flange. This height is used in the calculation of a concentric flange drop.

NextEra Evaluation

The current license basis postulated RVH drop considers a concentric and flat drop of the RVH from the maximum height allowed by plant procedures. In accordance with plant procedures, the RVH is moved vertically to its maximum height, before moving horizontally away from the reactor vessel.

NEI 08-05, Table 1 Initiative Analysis

If the analyses are based on an elastic-plastic curve, it must represent a true stress-strain relationship.

NextEra Evaluation

The BMI conduit analysis uses a true stress-strain relationship. The true stress-strain data was constructed using ASME Code minimum values.

NEI 08-05, Table 1 Initiative Analysis

The analysis will consider the "maximum damage" caused by the transfer of energy to the vessel and supports. Analysis that accounts for appropriate consideration of conservation of momentum is acceptable. It is also acceptable to consider dampening.

NextEra Evaluation

The evaluation of "maximum damage" is integral to the postulated RVH drop events defined in the current license basis. Using the maximum displacement from the current analysis as the force input for the analysis of effects on the BMIs considers the maximum damage to the BMIs.

A beta dampening value of 5% damping at 30 Hz was included in the models in accordance with NEI 08-05. Beta damping was used to assist in eliminating high frequency noise found in the response of the systems. The actual systems respond at approximately 17.24 Hz. Due to the linear behavior of beta damping, a damping value of approximately 2.875% will be experienced by the systems at the response frequency. This damping value will have a negligible effect on the actual response of the systems.

NEI 08-05, Table 1 Initiative Analysis

To overcome water leakage due to damage from a load drop, credit may be taken for makeup water for BWRs and borated water makeup for PWRs of adequate concentration that is required to be available by the technical specifications.

NextEra Evaluation

The RCS boundary was shown to remain intact. Therefore, makeup capacity is not pertinent to the analyses.

NEI 08-05, Table 1 Initiative Analysis

Only reactor vessel head drop is considered.

NextEra Evaluation

Only the RVH drop was considered in the current license basis RVH drop accident, and the resulting displacement time-histories calculated in the current license basis RVH drop analysis were applied to the BMI conduit models.

NEI 08-05, Table 1 Initiative Analysis

The analysis should include the weight of the reactor vessel (RV) head assembly below the hook.

NextEra Evaluation

Only the weight RVH assembly below the hook was considered in the current license basis RVH drop accident, and the resulting displacement time-histories calculated in the current license basis RVH drop analysis were applied to the BMI conduit models.

NEI 08-05, Table 1 Initiative Analysis

Area of consideration: Fall of the reactor vessel head from its maximum height allowed by plant procedures directly (concentrically or flat) on the vessel flange. In some plant procedures, the reactor vessel head may be moved horizontally and still be over the flange, and then lifted further. The maximum drop height is determined by the maximum height above the flange while the reactor vessel head center of gravity is still within the flange radius or over the flange. This height is used in the calculation of a concentric drop.

NextEra Evaluation

The current license basis postulated RVH drop considers a concentric and flat drop of the RVH from the maximum height allowed by plant procedures. In accordance with plant procedures, the RVH is moved vertically to its maximum height, before moving horizontally away from the reactor vessel.

NEI 08-05, Table 1 Initiative Analysis

The analysis will consider the actual medium controlled by plant procedures.

NextEra Evaluation

The current license basis RVH drop accident considered the actual medium controlled by plant procedures, and the resulting displacement time-histories calculated in the current license basis RVH drop analysis were applied to the BMI conduit models.

NEI 08-05, Table 1 Initiative Analysis

All components and structures in the load path for the reactor vessel head drop will be evaluated to assure deformation is limited, that the core remains covered and that cooling of the core is maintained.

NextEra Evaluation

Although BMI conduits are not in the impact load path, adherence with the approved guidance for the load path components provides consistency of methodology. Modeling both with and without potential conduit/floor interaction ensured the most limiting condition was evaluated. The BMI conduit model found the resultant stresses from the RVH drop would not exceed ASME Code allowable values, and the RCS fluid boundary would remain intact.

NEI 08-05, Table 1 Initiative Analysis

The RV head assembly should be considered rigid unless explicitly modeled. The deformation of components attached to the RV head may be realistically considered.

NextEra Evaluation

This initiative is contained within the postulated RVH drop events defined in the current license basis, and the resulting displacement time-histories calculated in the current license basis RVH drop analysis were applied to the BMI conduit models.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.71(e) requires that licensees shall periodically update their FSAR to assure that the information included in the report contains the latest information developed. This update shall contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement. The update shall also include the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request.

NUREG-0612 (Reference 6.4) presents an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads at nuclear power plants. On December 22, 1980, the NRC issued Generic Letter 80-113 which was supplemented on February 3, 1981, with the issuance of Generic Letter 81-07 regarding NUREG-0612 (Reference 6.4).

NextEra concludes that incorporation of the revised postulated reactor vessel head drop analysis into the licensing basis, due to the change in methodology of the analysis, requires a license amendment pursuant to 10 CFR 50.90.

4.2 Significant Hazards Consideration

The proposed amendment would revise Point Beach Nuclear Plant (PBNP) Final Safety Analysis Report (FSAR) Chapter 14.3.6, Reactor Vessel Head Drop Event, with information from a more recent analysis which has demonstrated that a postulated reactor vessel head (RVH) drop event would not result in the reactor coolant system (RCS) being breached, and therefore, no loss of RCS inventory would occur.

NextEra has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, Issuance of Amendment, as discussed below.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed amendment is limited in scope to a postulated RVH drop and the administrative controls in place, which limit the height of the reactor RVH lift, ensuring an actual drop is bounded by the analyses of record.

   Incorporation of the analysis performed in accordance with NRC-approved guidance, which demonstrates bottom-mounted instrumentation (BMI) conduits will not sever following a postulated RVH drop, does not increase the probability or consequences of a previously evaluated accident. The evaluation, in fact, demonstrates that if the postulated RVH drop occurred, the consequences would be significantly less than are now assumed because the ability to maintain a coolable geometry in the core has not been compromised. In accordance with the NRC-endorsed methodology contained in NEI 08-05 (Reference 6.1), which states, "Previous evaluations have indicated that the consequences of impacts between the upper vessel internals and the fuel were not significant with respect to public health and safety," a revised radiological analysis was not performed.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment is limited in scope to a postulated RVH drop and the administrative controls in place, which limit the height of the reactor RVH lift, ensuring an actual drop is bounded by the analyses of record.

Incorporation of the analysis performed in accordance with NRC-approved guidance, which demonstrates BMI conduits will not sever following a postulated RVH drop, does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment does not: (1) operate equipment in alignments or in a manner different from that previously evaluated in the FSAR; (2) install, remove or modify equipment important to safety; or (3) introduce new failure modes or effects for any existing system, structure or component.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment is limited in scope to a postulated RVH drop and the administrative controls in place, which limit the height of the reactor RVH lift, ensuring an actual drop is bounded by the analyses of record.

Incorporation of the analysis performed in accordance with NRC-approved guidance, which demonstrates BMI conduits will not sever following a postulated RVH drop, does not involve a significant reduction in a margin of safety. The evaluation, in fact, demonstrates that if the postulated RVH drop occurred, the consequences would be significantly less than are now assumed because the ability to maintain a coolable geometry in the core has not been compromised.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NextEra concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operating in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulation, and (3) the issuances of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The Plant Operations Review Committee has reviewed this amendment and concurs with this conclusion.

5.0 ENVIRONMENTAL CONSIDERATION

NextEra has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

(6.1) NRC letter to Nuclear Energy Institute, dated September 5, 2008, Safety Evaluation Regarding NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0 (ML082410532)

(6.2) NRC letter to Nuclear Management Company, LLC, dated September 23, 2005, Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Analysis Into the Final Safety Analysis Report (TAC Nos. MC7650 and MC7651) (ML052560089)

(6.3) NRC letter to Nuclear Management Company, LLC, dated January 12, 2006, Point Beach Nuclear Plant, Units 1 and 2 - Revision to Safety Evaluation for Amendment Nos. 220 and 226 (TAC Nos. MC7650 and MC7650) (ML052850005)

(6.4) U.S. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, dated July 1980 (ML070250180)

ENCLOSURE 1

ATTACHMENT 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 265
REVISION TO THE REACTOR VESSEL HEAD DROP METHODOLOGY

PROPOSED FINAL SAFETY ANALYSIS REPORT
CHAPTER 14.3.6 CHANGES
(MARK-UP)

21 pages follow

14.3.6 REACTOR VESSEL HEAD DROP EVENT

PBNP committed to incorporate an analysis of the Reactor Vessel Head (RVH) drop into the PBNP FSAR by letter NRC 2005-0094, dated July 24, 2005. The analyses presented in this section demonstrate that ~~the amounts of radioactivity released to the environment in the event of a postulated RVH drop event result in calculated offsite radiological doses that are well within the limits specified in 10 CFR 100, and result in calculated control room doses that do not exceed the limits of 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. The calculated doses are summarized in Table 14.3.6-3.~~ a limiting postulated RVH drop will not result in rupture of the RCS and associated pressure boundaries, and that the core will remain covered.

To resolve questions pertaining to a postulated RVH drop event initiated as a result of the Unit 2 reactor head replacement in 2005, analyses were performed and submitted for NRC review and approval. The analyses performed included two structural analyses that evaluated the effect of the impact on the impact load path through the reactor vessel, RCS piping, and the reactor vessel supporting structures. Included were radiological analyses predicated on an assumption that the impact would result in a clad gap release, and a presumptive failure of the bottom mounted instrumentation (BMI) conduits located beneath the reactor vessel. Reference 1 is the Safety Evaluation (SE) documenting NRC acceptance of those analyses and is applicable to both units. Subsequent analyses performed in accordance with later approved NRC methods, and utilizing the previously performed structural analyses, found that the BMI conduits would remain intact, and that the previous presumption of a clad gap release was not necessary. The maintaining of core cooling capability with normal decay heat removal, and the removal of the assumed clad gap release permitted the elimination of most of the additional ~~Point Beach made~~ regulatory commitments associated with the Reactor Vessel Head Drop Event (Reference 1 and Reference 2). Reference (13) is the Safety Evaluation (SE) documenting NRC approval of the analyses demonstrating that the BMI conduits would remain intact. It is applicable to both units. Reference (12) is the NRC endorsement of generic analytical approaches that discount a fission product gap release as a result of an RVH event. ~~Commitments pertaining to requirements to be met or in place prior to initiating a reactor vessel head lift are contained in Technical Requirements Manual (TRM) Table 3.9.4-1 "Required Administrative Controls During Reactor Vessel Head Lift."~~

14.3.6.1 Occurrences That Lead To The Initiating Event

While the potential causes of an RVH drop event are not specified in the NRC safety evaluations or the supporting submittals, such an event can be postulated to occur from mechanical failure of the crane hoist mechanism, cable failure, or RVH lift rig failure. The main hoist of each polar crane is equipped with two independent upper travel limit switches to prevent the possibility of a "two-blocking" incident. The two independent upper travel limit devices are of different design and are activated by independent mechanical means. These devices independently de-energize either the hoist drive motor or the main power supply. Since the upper travel limit switches on the containment polar cranes are independent, are tested, and operational restrictions limit upward travel, it was established in Reference 1 and Reference 3 that the potential for an RVH drop event due to "two-blocking" (i.e., exceeding the physical upper travel limits of the crane) is negligible. See FSAR Appendix A.3 for additional discussion on "two-blocking."

14.3.6.2 Event Frequency Classification

The initiating event in this assessment is the drop of the RVH while it is suspended over the reactor vessel. The RVH is assumed to fall onto the reactor vessel flange, resulting in damage to the reactor vessel support structure.

NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," was written to address NRC Candidate Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Crane operating history from 1968 through 2002 was reviewed as part of this report to provide a risk assessment associated with lifts of Very Heavy Loads (VHL). The risk analysis included in NUREG-1774 considers VHL lifts for any crane at any operating nuclear station. The analysis considers a postulated drop of load at any point during the movement of a load from the initial lift until set-down.

The probabilistic analysis contained within NUREG-1774 is primarily concerned with the probability of a VHL drop at an operating commercial nuclear power plant. A VHL is defined as any load over 30 tons. The generic probability for any VHL drop is given as 5.6E-5 per lift. This value is based upon three (3) drops per 54,000 VHL lifts.

Reference 1 and Reference 3 established that a postulated RVH drop meets the frequency classification of an infrequent incident (i.e., an incident that may occur during the lifetime of the plant).

14.3.6.3 Sequence of Events

The analyzed event is a concentric drop of the RVH onto the reactor vessel flange from a height of 26.4 feet. This was determined to impart the maximum credible impact loads on the reactor vessel and supporting structures. The resultant impact displaces the reactor vessel downward. Downward movement of the vessel creates the potential for damage to piping and tubing directly or indirectly connected to the reactor vessel, thereby creating a potential for a decrease in reactor coolant inventory.

Upon impact with the vessel flange, the kinetic energy of the vessel head is partially dissipated and partially transferred to both the head (rebound) and the vessel through an elastic/plastic collision. The impact forces, if high enough, can lead to yielding of the vessel supporting structures and/or attached piping.

After the head and vessel have come to rest, decay heat removal can be maintained by one or both RHR trains ~~and/or injection by the SI pumps. These same systems are adequate to makeup for any reactor coolant system (RCS) leakage that may occur from damaged Bottom Mounted Instrumentation (BMI) guide tubes.~~ Damage to the point of rupture or shearing of other connected piping, including the main RCS loops, pressurizer surge line, core deluge lines, accumulator dump lines, normal charging, BMI conduits, and cold leg SI Lines), etc. are not expected.

~~Should RCS leakage be substantial, RHR suction can be re-aligned to the accident sump to ensure a continued cooling and makeup once available suction sources are depleted.~~

~~The mechanical shock of the impact is also postulated to result in damage to all fuel assemblies which leads to a complete fission product gap release. The resulting release of radioactivity must be mitigated by containment to maintain control room and off-site exposures within acceptable limits. Establishing a stringent containment closure prior to lifting the RVH ensures that this mitigating function is maintained.~~

14.3.6.4 Plant Characteristics Considered in the Safety Evaluation

To demonstrate the capability of the reactor vessel, RCS, and supporting systems and structures to sustain a postulated RVH drop event, two complementary inelastic structure and piping system analyses were performed (Reference 6 and Reference 7). A RVH drop is postulated to occur during refueling when the head is manipulated above the reactor vessel. The RVH is assumed to fall concentrically onto the reactor vessel. Established administrative controls limit the maximum RVH drop height to 26.4 feet. This drop height has been utilized in the analyses discussed below. Both analyses were performed prior to NRC issuance of (but consistent with) Reference 12, which established approved methods for analyses of postulated RVH drop events.

The Sargent & Lundy (S&L) analysis (Reference 6) evaluated the reactor vessel and vessel support behaviors using a finite element model. The Westinghouse analysis (Reference 7) evaluated the plastic deformation that may occur to connected RCS piping based on specified bounding reactor vessel displacements.

S&L Finite Element Analysis

This analysis considers a flat vertical impact of the new RVH, using weights of 200,000 lbs for Unit 1 and 194,000 lbs for Unit 2, dropping from a height of 26.4 feet onto the reactor vessel flange. This analysis also includes an evaluation of the structural integrity of supporting elements in the load path, and predicts the vertical downward displacement of the reactor vessel.

The load path consists of the reactor vessel, reactor vessel supports at the four RCS nozzles and two brackets under the RHR core deluge nozzles, the support girder box frame, and the six pipe columns and their supports, which rest on the concrete foundation. The reactor coolant system (RCS) piping provides additional stiffness to the reactor vessel nozzles under vertical impact loading, and also transfers a portion of the impact load to the steam generator (SG) and the reactor coolant pump (RCP) support structures under a postulated RVH scenario. The concrete and embedded reinforcing bar located between the support girder and the concrete foundation under the support columns is not considered to provide any vertical support, even if the predicted deflection of the vessel could result in contacting the concrete.

The analysis models used are static analysis models for stiffness calculations of various components and substructures, and a dynamic impact model. The finite element analyses are performed using the ANSYS computer code.

The static analysis models include:

(1) A detailed model of reactor vessel flange and reactor vessel shell below the flange, including a nozzle resting on a supporting shoe.

(2) A similar detailed model of reactor vessel flange and reactor vessel shell below the flange with a support bracket resting on a supporting shoe.

(3) A detailed model of the hexagonal girder box frame supported by six pipe columns at the vertices.

(4) Piping models for the RCS hot legs and cold legs.

These models are used to construct static load-displacement diagrams for all steel components that are within the impact load path. Static vertical displacement is applied to the components uniformly and a reaction force is calculated to construct the force-displacement diagram of the affected components. In the static analysis, non-linear material properties are modeled with a strength increase factor of 10 percent to account for the strain rate effects due to the dynamic impact. The large deformation analysis option was selected to account for potential buckling and yielding in the structural components along the impact load path.

The results of the static analysis are used as part of the input for dynamic analysis. In calculating the stiffness of RCS hot leg or cold leg, two bounding cases are analyzed:

1) A fixed boundary condition is used at either the SG location or the RCP location.
2) A pinned boundary condition is used at the SG location or the RCP location.

In both cases, the pipe axial movement is released to account for the potential horizontal movement of the SG or the RCP.

The dynamic impact model consists of a two-mass model with springs and dash-pot in a vertical configuration. The top mass represents the falling head, and the bottom mass represents the target reactor vessel model supported by various springs, which represent the stiffness of the nozzle/bracket support, the girder box frame/column supports, and the RCS piping.

In the dynamic impact analysis, an impact damping of 5% of the critical damping is used. This assumption is judged to be reasonable for this application in consideration of:

1) energy loss due to plastic damage at the impact surface between the RVH and the reactor vessel flange;
2) energy loss due to imparted damage to six lateral supports for the hexagonal girder box frame; and,
3) energy loss due to local damage to the liner and concrete crushing at the top of the six support columns.

Results of the dynamic transient analysis for Unit 2 indicate that the maximum dynamic downward displacement of the reactor vessel is 2.72 and 3.20 inches for cases 1 and 2 respectively. These displacements are both less than the 3.375" necessary before the hexagonal girder box frame would come into contact with the concrete "shelf", and this is consistent with the assumption that the concrete shelf does not provide any resistance to downward motion.

Using the limiting downward displacement of 3.2", the maximum Von Mises stress in the nozzle due to membrane plus bending is less than the ASME Boiler and Pressure Vessel Code, Section III, Appendix F allowable stresses for membrane stress intensity of 0.7 S_u. Similarly, the Von Mises stress in the reactor vessel support brackets is also less than 0.7 S_u.

The S&L analysis also evaluated the maximum impact load on the column foundation, and the capability of the concrete shelf to provide lateral support for the stability of the support columns (i.e. to limit buckling) located within the shelf and found the results acceptable.

Westinghouse Plastic Analysis of RCS Loop Piping

The evaluation consisted of a plastic analysis of the PBNP reactor coolant loop piping for a downward vertical displacement of the reactor vessel nozzles. Two displacements were analyzed: (1) a 4-inch displacement, which bounds the displacement calculated by the S&L model, and (2) a 6.5-inch displacement, which represents the maximum possible displacement of the reactor vessel nozzles before the RCS piping comes in contact with the biological shield wall.

The results of the analysis were compared to the criteria specified in the 1998 Edition of ASME Code, Section III, Appendix F, Paragraph F-1340. The criteria allow for large RCS loop piping deformations, with the intent that violations of the RCS pressure boundary do not occur.

The analysis uses an ANSYS finite element model of the hot and cold legs. The hot and cold legs are fixed at both ends (the reactor vessel nozzles and the SG or RCP nozzles). Each leg was modeled as a straight run of piping with one elbow. The hot and cold leg material properties were represented by a piece-wise linear stress-strain curve. Two sets of material properties were used to represent the upper and lower bound properties of the piping and elbow materials.

The results of the analysis indicate that the maximum calculated stress intensity in the hot and cold leg piping is within the ASME Code, Section III, Appendix F limit of 0.7 S_u for general primary membrane stress for the 4-inch reactor vessel nozzle displacement. Since the 4 inch reactor vessel nozzle displacement bounds the maximum calculated vessel displacement predicted from the S&L model, there is reasonable assurance that the pressure boundary integrity of the RCS loop piping will be maintained in the event of a postulated RVH drop.

The results also indicate that the 0.7 S_u limit is exceeded for the cold leg for a 6.5-inch vessel nozzle displacement. The maximum stress intensity was calculated in the cold leg elbow. While the calculated stress intensity exceeds the ASME Code general primary membrane stress intensity limit, it is concluded that loss of the RCS piping pressure boundary integrity would not be expected even if the vessel nozzle displaced 6.5 inches. This is because the maximum calculated stress intensity is still well below the material ultimate strength.

<u>Analysis of Reactor Vessel Deflection</u>

Based on the Sargent & Lundy FEA provided in Reference 6 and the Westinghouse analysis provided in Reference 7, the following bounding conditions apply:

Following the postulated RVH drop, using a conservatively estimated RVH weight of 200,000 lbs (Unit 1), the reactor vessel deflection would not exceed 3.36 inches. This calculated deflection is slightly greater than the Unit 2 calculated vessel deflection due to the conservative weight assumed and slight dimensional differences between units. RCS piping remains intact following the postulated reactor vessel deflection.

The impact of the postulated reactor vessel deflection on the attached RCS piping was assessed. This assessment was performed by Westinghouse and is documented in Reference 7. Westinghouse performed an analysis for a 4-inch deflection, which bounds the projected reactor vessel deflection. The results of the analysis show that stress values are less than the more restrictive criteria of 0.7 S_u specified in ASME Section III Appendix F. In addition, a second case to analyze a deflection value of 6.5 inches, which is equivalent to the gap that exists between the RCS piping and the shield wall, was conducted. The results of this analysis yielded stress values of greater than 0.7 S_u but did not predict failure of the RCS piping.

The combined results of the Sargent & Lundy and the Westinghouse analyses show that the damage from a RVH drop would not result in a loss of decay heat removal. Based on these results, it was concluded that adequate reactor core cooling and makeup capability would be maintained following the expected deflection of the reactor vessel from a postulated RVH drop.

Piping attached to the reactor coolant system (RCS) was not modeled or specifically analyzed for deflection and stress values as a result of the vessel deflection from a RVH drop. Based on the ability to analyze and demonstrate RCS piping acceptability for a bounding deflection of 4 inches, it was determined that the attached piping would also be acceptable. This conclusion was based upon the fact that all connections to the RCS piping are outside of the biological shield wall; thus, the deflection would be much less than the total deflection of the RCS piping. In addition, the attached piping is of smaller diameter and is more flexible. The main connections to the RCS, credited for maintaining core cooling and makeup following a RVH drop, are the residual heat removal (RHR) lines, cold leg safety injection (SI) injection lines and charging. The RHR suction and return lines are 10-inch lines; the cold leg SI flow path is through the 10-inch SI accumulator injection line connected to the RCS.

Charging and auxiliary charging are connected through a 3-inch and 2-inch line to the RCS. The 10-inch connections are the closest connections of concern to the reactor vessel, with one exception, and would therefore experience the greatest relative deflection. The only exception is that the Unit 1 Auxiliary Charging line is 10 inches closer to the reactor vessel than the corresponding Safety Injection line on the "B" cold leg. Since the Auxiliary Charging line is a 2-inch line with greater flexibility than the 10-inch SI line, the focus was on addressing the SI lines. For Unit 1, the ratio of the distance from the reactor vessel to the steam generators or reactor coolant pumps would yield a deflection of approximately 20 percent, or less, of the total vessel deflection. For a vessel deflection of 3.36 inches, the deflection at the connection would be approximately 0.67 inches.

In Unit 1, the shortest horizontal piping run from the 10-inch connections at the cold legs to the first vertical support (which is a spring hanger), is greater than 6 feet. The shortest vertical run is approximately 10 feet (on the opposite cold leg). Both connections have horizontal offsets that decrease their stiffness in the vertical direction. The shortest horizontal run to an anchor is greater than 14 feet with an intervening vertical loop.

The RHR return line connects to the SI accumulator injection line over 22 linear feet from the B loop cold leg connection. The condition is very similar for the RHR suction line connection to the A hot leg. The distance to the closest anchor is greater than 13 feet with an intervening vertical loop containing an additional 30 feet of piping.

In each case, the total linear distance between anchors for the attached piping is greater than the worst RCS piping case, and that case was shown to be acceptable for a deflection of 4 inches. Based on this, the added flexibility of smaller diameter piping and an equivalent deflection of approximately 0.67 inches, it was determined that a detailed analysis of the connected piping was not necessary.

Additionally, the integrity of the two 6-inch core deluge lines was evaluated based on comparing the section properties and applicable pipe spans to the RCS piping. This comparison, coupled with the fact that the core deluge lines are more flexible than the RCS piping, leads to the conclusion that the integrity of the core deluge lines are bounded by the assessment for the RCS piping.

Bottom-Mounted Instrument (BMI) Tubes

Reference (14) analyzed the stresses of the BMI conduits that result from the maximum downward displacement of the reactor vessel. The analysis was performed in accordance with the NRC-approved guidance of Reference (12) and concluded that 0.7 S_u would not be exceeded in any of the BMI conduits. Therefore, no loss of integrity of the BMI conduits is expected, and the RCS inventory would be retained. ~~As a result of the predicted maximum dynamic downward displacement of 3.2 inches for the reactor vessel, and recognizing the potential impact between the BMI tubes and the floor (clearance varying between 1" and 4.5"), it was conservatively assumed that all 36 tubes are severed. Therefore, the structural integrity of the BMI tubes is not considered in the structural integrity analysis.~~

~~14.3.6.5 Protective System Actions~~

~~Core Cooling Configuration~~

~~The emergency core cooling system (ECCS) and normal core injection paths remain available during the postulated event. Core cooling water remains available to ensure adequate cooling and makeup is maintained to remove decay heat and keep the core covered.~~

~~Upon exhaustion of the Refueling Water Storage Tank (RWST) inventory, the residual heat removal (RHR) pumps would be realigned to take suction from the containment sump; with the safety injection pump(s) drawing from the RHR pump discharges as needed. This provides assurance that core cooling and makeup can be maintained for a prolonged period.~~

~~For Low Temperature Overpressure Protection of the reactor vessel, Limiting Condition for Operation (LCO) 3.4.12 requires one train of safety injection to be disabled when the RVH is installed on the vessel. Prior to head installation, one train of safety injection is configured to prevent inadvertent start and pressurization once the RVH is installed in order to satisfy this requirement. PBNP will maintain the second train as available using administrative controls currently defined in the shutdown safety assessment procedure. This procedure defines the safety assessment and risk management process used to comply with 10 CFR 50.65(a)(4) of the Maintenance Rule. In order to maintain the second train as available during head lift operations, PBNP has specified administrative requirements which will ensure prompt recovery if required. Pre-briefed and stationed operators would take the necessary local manual actions to ensure a timely recovery of the second train as necessary.~~

~~In the event of a RVH drop, operators will first be alerted by reports from personnel who witnessed the event. A Senior Reactor Operator (SRO) will be stationed inside containment during all such lifts. The function of the SRO is to communicate the occurrence of a RVH drop to the control room.~~

~~The operators will assess damage using indications available to them in the control room. The mitigating strategy will retain core cooling and makeup using RHR, charging and safety injection flow paths. Minimum equipment availability is established as prescribed in the regulatory commitments listed at the beginning of FSAR 14.3.6. Based on the need for an assured makeup capability, both trains of RHR and SI are required to be operable and available, respectively, during RVH lifts above a vessel containing irradiated fuel.~~

~~As an upper bounding scenario, all bottom mounted instrument (BMI) tubes are postulated to sever. The resultant inventory loss from the RCS is postulated to be approximately 300 gpm. This value is based on all 36 BMI tubes being severed in a manner that causes minimal flow restriction from the break and the resultant gravity fed flow of RCS water through the tubes' nominal interior diameters. The boiloff rate is minor when compared to the loss because complete failure of all 36 BMI tubes and is well within the capacity of a single SI or RHR pump. (Each RHR pump is rated for 1560 gpm at a design head of 280 ft, and an SI pump is rated for 700 gpm at a head of 2600 ft).~~

~~In the event of a loss of coolant event during cold shutdown, procedure controls are established to ensure adequate core inventory and cooling are maintained. The procedure performs the following:~~

~~1) Check RHR Pump conditions and secure, if required, due to system voiding~~
~~2) Establish charging from the RWST to maintain RCS inventory~~
~~3) Establish safety injection flow from the RWST via one SI pump, if necessary~~
~~4) Verify adequate injection flow based on inventory and RCS temperature~~
~~5) Establish low head injection flow, if required, based on temperature~~
~~6) Establish containment sump recirculation, if required, based on RWST level.~~

~~Time to boil curves provided in SEP-1, "Degraded RHR System Capability," show that the time to boil, 100 hours after shutdown as assumed in the current dose assessment, with RCS level at reduced inventory and starting at 140°F, is approximately 18 minutes. With the RCS level at one foot below the flange, which is the procedural requirement for lifting and setting the RVH and the same conditions as above, the time to boil is just over 23 minutes. Based on simulator validation of steps in SEP 2.3, "Cold Shutdown LOCA," the time to inject water into the RCS using SI pumps is approximately 10 minutes. Thus, adequate time is available to diagnose the problem, enter the associated shutdown emergency procedure, and establish makeup flow following a postulated RVH drop that results in RCS leakage.~~

~~Upon exhaustion of the minimum committed 243,000 gallon suction source (nominally the Refueling Water Storage Tank, RWST)[1], the RHR pump(s) can be realigned to take a suction from the containment sump. Accordingly, this accident sump suction path must be maintained available during movement of the RVH.~~

~~In addition to the provision of in-depth makeup capability, containment is required to be closed with the purge supply and exhaust fans off and the associated penetrations closed (either by valves or their equivalent), and personnel airlock door interlocks are functional (to ensure that at least one door at each airlock is closed) prior to movement of the RVH.~~

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1 ~~This suction source is 243,000 gallons available for recirculation. This would be in addition to an inventory necessary to fill the lower refueling cavity. This is because the volume in the lower refueling cavity is isolated from the containment sump during head lift activities, and would not be available for dilution and hold-up of the source term inventory. Therefore, if the lower cavity is not flooded at the time of the lift, the available suction source must be 243,000 gallons *plus* the volume needed to fill the lower cavity.~~

~~14.3.6.6 Core and System Performance~~

~~The postulated RVH drop occurs with the reactor in cold shutdown with shutdown margin assured by a combination of inserted control rods and boron concentration. The drop is not postulated to result in changing the core geometry, and as such, no explicit evaluation or analysis of nucleonics has been performed.~~

~~System performance is addressed in Section 14.3.6.5 (Protective System Actions) above.~~

~~14.3.6.7 Barrier Performance~~

~~The primary barrier against the adverse consequences of an RVH drop is prevention. The measures in place to minimize occurrence of such an event are described in detail earlier in this section.~~

~~The fuel pins are suspended by friction in the grid strap assemblies, and are relatively stiff in the axial direction. In addition, there is some axial spacing between the upper fuel nozzle block and the ends of the fuel pins. As a result, the pins can slide axially to accommodate some axial shock loading of the fuel assembly. Therefore, fuel clad failure is not anticipated to occur due to an RVH drop event. Nonetheless, to provide a bounding accident scenario, a failure of 100% of the fuel cladding is assumed by the radiological dose assessment. This assumed failure releases all of the clad gap inventory, but since the fuel remains covered, further degradation (e.g., Zr-water reaction, fuel melt) does not occur, and the ceramic fuel is assumed to retain all additional radionuclide inventory.~~

~~As previously discussed, a gross failure of the RCS pressure boundary is not expected. Some leakage is assumed to occur due to the assumed severance of BMI tubes. This leakage is well within the makeup capability of the SI and/or RHR systems, and this prevents further fuel damage due to uncovery.~~

~~Lastly, containment closure is a pre-condition for movement of the RVH over irradiated fuel (an exception is when the head is less than 24" above the vessel flange, when opening of the purge supply and exhaust valves and initiation of purge is permitted). As such, the containment comprises the final barrier against release of radioactive materials.~~

~~14.3.6.8 Radiological Consequence~~

~~An evaluation was performed to assess the offsite and control room dose consequences following a RVH drop for either unit. The event sequence assumes that the RVH drops onto the vessel causing fuel cladding damage to all of the fuel assemblies in the core, which results in a gap release. In addition, damage to the BMI tubes is assumed such that coolant is lost through these penetrations. Initial makeup of the RCS to the vessel is via suction from the RWST to the safety injection pumps, RHR pumps, or charging pumps. Once the RWST volume is exhausted, the RHR system is realigned to recirculate the coolant in the containment sump to maintain the core sub-cooled.~~

~~The dose evaluation for the RVH drop accident scenario described above uses a combination of the input assumption guidance for a design basis loss-of-coolant accident (LOCA) and a fuel handling accident (FHA), as they apply to the accident scenario. There is currently no explicit~~

~~guidance for assessing the dose consequence of the RVH drop (or any heavy load drop) that results in an uncontrolled loss of coolant in addition to fuel damage.~~

~~For purposes of providing a bounding source term, the RVH drop is assumed to result in clad damage to 100% of all fuel assemblies, such that a complete gap release occurs.~~

~~Prior to moving the RVH, the following conditions are assumed to be in effect:~~

- ~~Reactor has been shut down for a minimum of 100 hours;~~
- ~~Containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks functional);~~
- ~~Purge supply/exhaust system fans are off and isolation valves closed;~~
- ~~Other containment penetrations that allow containment atmosphere to communicate with the environment or the PAB atmosphere are closed.~~

~~There are two possible release paths for a postulated RVH drop at PBNP:~~

- ~~Containment - Through open penetrations or leakage through containment barriers,~~
- ~~ECCS leakage - Due to eventual operation of the recirculation mode of the RHR system.~~

~~Since the release paths considered for the RVH drop are the same as those for the FSAR 14.3.5 LOCA radiological analysis, the dose consequences post-RVH drop are determined by scaling the FSAR LOCA dose consequences for the RVH drop source term and sump volume (dilution). Additionally, the control room operator doses are scaled to account for a higher unfiltered in-leakage into the control room and an assumed increased emergency core cooling system (ECCS) leakage rate. The scaling factor for each input assumption is determined by a ratio of the RVH drop input assumption and LOCA dose input assumption. The input assumption scaling factors are applied to the LOCA dose values to estimate the RVH drop dose consequence.~~

~~These release paths were evaluated using the guidance provided in Regulatory Guide (RG) 1.195. The guidance is appropriate for PBNP since 10 CFR 100 is the licensing basis for all of the design basis accident analyses except the fuel handling accident, which is licensed to 10 CFR 50.67. (NRC safety evaluation dated April 2, 2004, Reference 8.)~~

~~Acceptance Criteria~~

~~The guidance in RG 1.195 does not provide specific accident offsite dose consequence criteria for the heavy load drop event. However, NUREG-0612 does categorize the heavy load drop event to be in the same class of limiting faults for which the radiological dose acceptance criteria are stated to be "well within" 10 CFR 100, meaning 25% of the 10 CFR 100 limits. The FHA is within this class of accidents, which generally would also be used to assess a heavy load drop such as RVH. Therefore, the offsite dose criteria are 6.3 rem whole body and 75 rem thyroid as listed in Table 4 of RG 1.195. The PBNP control room dose criteria are 5 rem whole body, and 30 rem thyroid as provided by 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 and clarified by NUREG-0800 Section 6.4.~~

~~Containment Release Evaluation~~

~~The RVH drop dose assessment assumes that containment closure is established prior to occurrence of the event. Containment closure is defined as the containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks functional), the purge supply/exhaust system fans off and the isolation valves closed, and all penetrations that allow containment atmosphere to communicate with outside or PAB atmosphere closed.~~

~~The postulated RVH drop does not result in pressurization of the containment building. This is because the event occurs while the reactor coolant system is open to the containment building atmosphere and sufficient RCS makeup is available to provide cooling to the core. Heat is removed from containment by makeup to the RCS and containment sump recirculation. Since the containment configuration meets Footnote 2 to Position 5.1 of Appendix B to RG 1.195, and there is no pressure differential induced by the accident, there is no release via containment leakage.~~

~~ECCS Leakage Release Path Evaluation~~

~~The radiological dose assessment for the RVH drop assumes that the impact of the RVH onto the vessel results in cladding damage to 100% of the fuel assemblies. The event scenario also assumes complete severance of the BMIs such that there is an uncontrollable loss of coolant. Following confirmation of the event by Operations, makeup to the RCS is initiated to ensure that the core is covered and sub-cooled. Just prior to the exhaustion of the RWST inventory, the RHR pumps would be realigned to take suction from the containment sump; with the SI pump(s) drawing from the RHR pump discharges as needed. This provides assurance that core cooling and makeup can be maintained for a prolonged period. It is assumed that during the recirculation phase of the accident, this system is leaking containment sump coolant at a rate twice the limits of the Leakage Reduction and Preventive Maintenance Program for the emergency core cooling system (ECCS). No release during the injection phase is assumed.~~

~~Although the RVH drop results in a LOCA, the damage to the fuel assemblies is not driven by a thermal-hydraulic event but is assumed to occur due to impact of the head. This event occurs after the reactor has been shutdown and while the reactor vessel is open to the atmosphere. The accident scenario assumes that operations responds to an RVH drop by initially injecting borated coolant and eventually placing the RHR system in containment sump recirculation mode to provide long term decay heat removal. Since the accident occurs at temperatures and pressures well below operating levels and the accident mitigation strategy ensures that the core is covered and cooled, no additional fuel damage would occur. Therefore, the non-LOCA gap fractions provided in Table 2 of RG 1.195 are applicable to the RVH drop event since no additional release from the fuel (e.g., due to fuel melt) will occur. RG 1.195 provides a larger gap fraction for I-131 than for the other isotopes of iodine. Therefore, this evaluation conservatively applied the I-131 gap fraction of 8% to all isotopes of iodine.~~

~~The amount of activity released from the gap is determined from the total core inventory assumed for the LOCA analysis adjusted for the decay time of 100 hours from shutdown and the nuclide gap release fractions. The LOCA core inventory at shutdown can be found in PBNP FSAR Table 14.3.5-1. Font. For the ECCS leakage path, all of the gap activity of iodine is assumed~~

~~to be retained in the coolant while the noble gases are not retained in any appreciable amount in the coolant. Therefore, consistent with RG 1.195, Appendix B, Position 3, the evaluation did not consider a noble gas release through ECCS leakage. The source term for the RVH drop in the coolant (i.e., the sump source term) is provided in Table 14.3.6-1 below. The LOCA sump source term, which is one-half the total core inventory, is also provided. The source terms are based on power operation at 1549 MWt (1540 MWt licensed power plus 9 MWt calorimetric uncertainty).~~

~~The amount of coolant available for recirculation is equal to the amount of coolant that is injected. It is assumed that 243,000 gallons of borated coolant is injected into the vessel. The volume of coolant initially in the vessel and RCS is not credited for determining the dose consequence. The sump volume credited in the RVH drop analysis is larger than the value assumed in the LOCA analysis. A detailed discussion regarding the impact of increase in volume on the dose consequences is provided below.~~

~~The offsite and control room doses due to the RVH drop ECCS leakage are estimated by adjusting the LOCA ECCS leakage doses for the resulting RVH drop source term (Ci) and sump volume (gallons). Additionally, the control room dose estimates for the RVH drop take into account the ECCS leakage rate factor of two multiplier (Position 4.2 of Appendix A of RG 1.195) and a higher unfiltered in-leakage rate. The offsite LOCA ECCS leakage doses already take into account the factor of two multiplier for the ECCS leakage rate and are not impacted by unfiltered in-leakage. No other adjustments are made on LOCA ECCS leakage doses, other than those discussed above. Therefore, all other input assumptions used in the LOCA radiological consequence analysis are, in effect, used in the RVH drop ECCS leakage analysis.~~

~~Since dose is directly proportional to the amount of activity present, scaling factors can be used to estimate the dose due to RVH drop ECCS leakage. The scaling factors for the source term is listed in Table 14.3.6-2 and calculated by dividing the LOCA ECCS sump source term by the RVH drop sump source term found in Table 14.3.6-1.~~

~~The scaling factor for the source term accounts for the change in total activity present in the sump coolant as compared to the LOCA analysis. The I-131 sump source term scaling factor is used to adjust the dose due to the fact that it minimizes the dose reduction.~~

~~The activity in the sump is released to the environment via leakage from the ECCS during the containment sump recirculation phase of the accident. The activity released over time, post-sump recirculation (Ci), is directly proportional to the sump concentration (Ci/cc) multiplied by the leakage rate (cc/min). Increasing the sump volume reduces the sump concentration, which directly reduces the activity released. The scaling factor representing the increase in sump volume available for recirculation is calculated by dividing the LOCA sump volume of 197,000 gallons by the RVH drop sump volume of 243,000 gallons. The scaling factor for the sump volume accounts for the change in total activity released to the environment as compared to the LOCA analysis.~~

~~The control room operator doses are further scaled to account for an increase in unfiltered in-leakage and an assumed increase in the ECCS leakage rate. In order to assess the impact of the measured unfiltered in-leakage value on the control room operator dose post-RVH drop, a~~

~~review of the constituents of the control room dose via the ECCS leakage pathway for PBNP is provided. The control room operator inhalation and whole body dose (due to activity internal to the control room) for PBNP is driven primarily by the amount of activity that passes through the control room ventilation filter into the control room. As described in FSAR Section 9.8, the control room emergency filtration system (CREFS) is actuated by a high radiation signal from control room area monitor RE-101, a high radiation signal from noble gas monitor RE-235 (located in the supply duct to the control room), or manually from panel C67. When the CREFS is in operation (referred to as MODE 4), 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 iodine and 99% for particulate iodine, the activity is entering the control room via the ventilation system at an estimated rate of 250 cfm for elemental/organic iodine and 50 cfm for particulate iodine. The current licensing basis control room habitability analysis described in FSAR 14.3.5 assumes that when CFEFS is in MODE 4, the unfiltered in-leakage is 10 cfm. See Section 9.8.3 for a discussion of the current status of an operable but non-conforming condition identified for the control room emergency filtration system and unfiltered in-leakage.~~

~~The LOCA assumes that CREFS is actuated by a high radiation signal at the onset of the accident due to the magnitude and timing of the release from containment since the recirculation of the containment sump does not occur until 20 minutes post-accident. The delay in the start of sump recirculation is also true for the RVH drop but automatic actuation of CREFS as a result of containment leakage will not occur. However, there is sufficient time between accident recognition and release initiation to credit manual actuation of CREFS, in addition to, the magnitude of the activity released to the environment is large enough to ensure CREFS actuation via a high radiation signal. Therefore, no delay in CREFS actuation is taken into consideration for the RVH drop control room dose assessment. The assumption that CREFS is operating simultaneously at the initiation of recirculation of the containment sump is maintained for the RVH drop control room dose assessment.~~

~~The LOCA ECCS leakage pathway is assumed to contain only elemental iodine. Therefore, under the current licensing basis analysis, the dose to the operator from radioactivity internal to the control room 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 (Reference 10). Incorporation of the measured unfiltered inleakage value into the RVH drop dose assessment results in an increase that is proportional 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 iodine dose increases by a factor of 1.35 (350 cfm / 260 cfm). Since the ECCS leakage pathway is assumed to contain only elemental iodine activity, the control room doses increase at most by a factor of 1.35.~~

~~The offsite RVH ECCS leakage doses are conservatively estimated by dividing the LOCA ECCS leakage dose values by the I-131 scaling factor. The control room external and internal cloud RVH ECCS leakage doses are conservatively estimated by dividing the LOCA ECCS leakage dose values by the I-131 scaling factor then multiplying by 2.7, accounting for the ECCS leakage rate multiplier and the measured unfiltered inleakage. Including the impact of~~

~~the measured unfiltered inleakage on the external cloud whole body dose is conservative, since this assumption does not actually impact the external cloud dose.~~

~~Table 14.3.6-3 below provides the LOCA dose values and the scaled dose values for the RVH drop event. The LOCA ECCS leakage doses are taken from FSAR Table 14.3.5-6, except for the "control room (external)" values. The external cloud control room dose is discussed in FSAR Chapter 11.6 and represents the contribution to the control room operator whole body dose due to the activity external to the control room from release pathways, containment and ECCS leakage. FSAR 11.6 Reference 2 provides the pathway breakdown of the external cloud dose and was used to determine the ECCS leakage contribution to the external cloud dose. Per Reference 2 of FSAR 11.6, the dose value at 5 feet from the shielded control room window is 0.20 rem and at 10 feet from the shielded control room window is 0.12 rem.~~

~~The value for control room (external) provided in Table 14.3.6-3 was calculated by summing the location doses (5 feet and 10 feet from shielded control room window) weighted by the time spent in front of the window (25% occupancy at 5 feet and 75% occupancy at 10 feet). Since these values are derived for the design basis LOCA, the accident duration is taken to be 30 days.~~

Conclusion

In the event of a worst case postulated RVH drop, the RCS pressure boundary remains intact and the core remains covered. There would be no loss of RCS inventory, and no release of fission products from the reactor core. As such, no extraordinary measures are necessary to mitigate the consequences of a postulated RVH drop. Administrative controls limit the height of a reactor vessel head lift, ensuring that any real drop is bounded by the analyses of record. ~~As seen in Table 14.3.6-3, the site boundary and low population zone RVH ECCS leakage thyroid and whole body doses are less than the acceptance criteria of 75 rem thyroid and 6.3 rem whole body. The control room doses are less than the acceptance criteria of 30 rem thyroid and 5 rem whole body. Therefore, the bounding dose consequence of the RVH drop can be considered "well within" the 10 CFR 100 limits. The current licensing basis radiological consequence of a design basis LOCA bounds the dose consequence of the postulated RVH drop, and the LOCA remains the limiting event for control room habitability. The source term assumed for the RVH drop event provides a significant margin of safety.~~

14.3.6.9 References

1. NRC Safety Evaluation, Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis Into the Final Safety Analysis Report," September 23, 2005, as revised by NRC letter "Point Beach Nuclear Plant, Units 1 and 2 - Revision to Safety Evaluation for Amendment Nos. 220 and 226 (TAC Nos. MC7650 and MC7651," dated January 12, 2006.

2. Point Beach Letter from D. L. Koehl to NRC, "Request for Review of Heavy Load Analysis," NRC 2005-0094, July 24, 2005.

3. NRC Safety Evaluation dated June 24, 2005, Point Beach Nuclear Plant, Unit 2 - Issuance of Amendment Re: Incorporation of Reactor Vessel Head Drop Accident Analysis Into the Final Safety Analysis Report," and revised by letter dated August 11, 2005 "Point Beach Nuclear Plant, Unit 2 - Revision to Safety Evaluation for Amendment No. 225."

4. NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," July 2003.

5. NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3.

6. Sargent & Lundy Calculation 2005-06760, Rev. 3, "Analysis of Postulated Reactor Head Load Drop Onto the Reactor Vessel Flange," July 22, 2005.

7. Westinghouse Calculation Note CN-RCDA-05-68 Rev. 2, "Plastic Analysis of Point Beach Reactor Cooling Piping for Reactor Vessel Head Drop," July 21, 2005.

8. NRC Safety Evaluation dated April 2, 2004, "Point Beach Nuclear Plant, Units 1 And 2 - Issuance of Amendments Re: Technical Specification 3.9.3, Containment Penetrations, Associated With Handling Of Irradiated Fuel Assemblies And Use Of Selective Implementation of the Alternative Source Term For Fuel Handling Accident."

9. NRC Regulatory Guide 1.195, "Methods And Assumptions for Evaluating Radiological Consequences Design Basis Accidents At Light-Water Nuclear Power Reactors," May 2003.

10. Point Beach Letter from A.J. Cayia to NRC dated December 5, 2003 (NRC 2003-0116), "Generic Letter 2003-01, Control Room Habitability - Response To Commitments."

11. Calculation 2005-0033, Rev. 0, "Dose Consequences Post-Reactor Vessel Head Drop at Various Times Post-Shutdown."

12. NRC SE dated (________), Point Beach Nuclear Plant Units 1 and 2, Issuance of Amendments, Revision to Reactor Vessel Head Drop Methodology, Amendment Nos. XXX and XXX.

13. NRC Letter to Nuclear Energy Institute, dated September 5, 2008, Safety Evaluation Regarding NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0, (ML082410532)

14. Westinghouse Calculation Note CN-MRCDA-0851, Revision 1-A, "Evaluation of Bottom Mounted Instrumentation (BMI) Conducts for a Postulated Closure Head Assembly Drop Event, dated December 4, 2009.

~~Table 14.3.6-1 LOCA AND RVH SUMP SOURCE TERMS~~

~~Nuclide~~	~~FSAR 14.3.5 LOCA Sump Source Term (Ci)~~	~~RVH Drop Sump Source Term (Ci)~~
~~I-131~~	~~2.07E+07~~	~~2.36E+06~~
~~I-132~~	~~2.96E+07~~	~~1.99E+06~~
~~I-133~~	~~4.23E+07~~	~~2.46E+05~~
~~I-134~~	~~4.65E+07~~	~~0.00E+00~~
~~I-135~~	~~3.95E+07~~	~~1.77E+02~~

~~Table 14.3.6-2 SUMP SOURCE TERM SCALING FACTOR~~

~~Nuclide~~	~~Scaling Factor~~
~~I-131~~	~~8.8~~
~~I-132~~	~~14.9~~
~~I-133~~	~~172.0~~
~~I-134~~	~~-~~
~~I-135~~	~~2.23E+05~~

~~Table 14.3.6-3 RVH DROP DOSE CONSEQUENCE (rem)~~

	~~FSAR 14.3.5 LOCA ECCS Leakage (rem)~~		~~RVH Drop ECCS Leakage (rem)~~	
~~Location~~	~~Thyroid~~	~~Whole Body~~	~~Thyroid~~	~~Whole Body~~
~~Exclusion Area Boundary~~	~~57.12~~	~~0.24~~	~~5.3~~	~~0.022~~
~~Low Population Zone~~	~~37.0~~	~~1.06~~	~~3.4~~	~~0.006~~
~~Control Room~~	~~106.7~~	~~1.144~~	~~26.5~~	~~0.04~~