NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

September 7, 2005

NRC 2005-0117
10 CFR 2.390

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

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

Request for Withholding of Proprietary Information from Public Disclosure

References:
1. Letter from NMC to NRC dated May 13, 2005 (NRC 2005-0063)
2. Letter from NMC to NRC dated June 9, 2005 (NRC 2005-0072)
3. Letter from NMC to NRC dated June 20, 2005 (NRC 2005-0079)
4. Letter from NMC to NRC dated July 24, 2005 (NRC 2005-0094)
5. Letter from NMC to NRC dated July 1, 2005 (NRC 2005-0084)

In Reference 1, Nuclear Management Company, LLC (NMC), submitted Westinghouse Report, "Assessment of Reactor Vessel Head Drop, Point Beach Unit 1 and Unit 2", (LTR-RCDA-05-428, Revision 1, dated May 13, 2005). Reference 2 submitted Sargent & Lundy Calculation, "Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange", dated June 8, 2005, and Automated Engineering Services Corporation Calculation, "Finite Element Analysis of Reactor Vessel Head Drop on Reactor Vessel Flange", dated June 8, 2005. Reference 3 submitted Sargent & Lundy Report, "Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange", Revision 1, dated June 19, 2005. Reference 4 submitted Sargent & Lundy Report, "Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange", Revision 3, dated July 22, 2005. Reference 5 submitted Westinghouse letter WEP-05-168, Revision 1, dated June 20, 2005, containing Westinghouse Report, "Point Beach Unit 1, Cycle 29 – Reactor Vessel Upper Closure Head Volume Best-Estimate Mean Fluid Temperature", (LTR-RCDA-05-379, Revision 1). All these submittals contained proprietary information.

Westinghouse Electric Company ("Westinghouse"), the owner of the information, has informed NMC that LTR-RCDA-05-428, Revision 1, which was submitted as a proprietary document in Reference 1, was reclassified by Westinghouse as non-proprietary. Enclosure 1 provides Revision 2 of this document, which is non-proprietary and may be released for public disclosure by the NRC. No changes to the content of this document were made.

APO1

References 2, 3 and 4 contained the same set of certified material test reports (CMTRs) for the Point Beach Unit 2 reactor vessel. These CMTRs contain information that is proprietary to Westinghouse. The specified pages (marked "Proprietary") of the documents containing the CMTRs were initially provided in Reference 2 and are supported by the authorization letter and affidavit submitted therein. Enclosure 2 to this letter provides a non-proprietary version of the CMTRs.

Westinghouse has reissued LTR-RCDA-05-379, Revision 1 (submitted in Reference 5), in both proprietary and non-proprietary versions. Enclosure 3 provides the reissued Westinghouse reports, "Point Beach Unit 1, Cycle 29 – Reactor Vessel Upper Closure Head Volume Best-Estimate Mean Fluid Temperature", dated August 30, 2005, (Proprietary) and "Point Beach Unit 1, Cycle 29 – Reactor Vessel Upper Closure Head Volume Best-Estimate Mean Fluid Temperature", dated August 30, 2005 (Non-proprietary). Also provided in Enclosure 3 are a Westinghouse authorization letter, accompanying affidavit, Proprietary Information Notice and Copyright Notice for the reissued reports.

Since the document listed above as Proprietary contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity, for each, the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the above documents, or the supporting Westinghouse affidavit, should reference the appropriate authorization letter (CAW-05-2050) and be addressed to B. F. Maurer, Acting Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 7, 2005.

Dennis L. Koehl

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

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

ENCLOSURE 1

WESTINGHOUSE REPORT,
"ASSESSMENT OF REACTOR VESSEL HEAD DROP, POINT BEACH UNIT 1 AND UNIT 2",
(LTR-RCDA-05-428, REVISION 2, DATED SEPTEMBER 1, 2005)
(NON-PROPRIETARY)

(16 pages follow)

Westinghouse

To: Kerry B Hanahan
cc: David H. Roarty
John Ghergurovich

Date: 9/1/2005

From: Reactor Component Design & Analysis
Ext: 412-374-6253
Fax: 412-374-6647

Your ref:
Our ref: LTR-RCDA-05-428, Rev. 2

Subject: **Assessment of Reactor Vessel Head Drop, Point Beach Unit 1 and Unit 2**

Ref. 1 WEP-82-584, "Wisconsin Electric Power Company, Point Beach Nuclear Plant, Reactor Vessel Head Drop Analysis," W. J. Johnson, November 15, 1982, (Westinghouse Proprietary).
Ref. 2 CN-RCDA-05-46, Rev. 2, "Comparison of Original and Replacement Head Drops for Point Beach Unit 1 and Unit 2," George J. Demetri, May 2005, (Westinghouse Proprietary).

At the request of Mr. Joe McNamara, Point Beach Engineering, Westinghouse's Mr. David Roarty recently performed an analysis of the Point Beach Reactor Vessel Head Drop Analysis (Ref. 1) to confirm its accuracy and adjust for the different weights of the new RRVH/HAUP package. Mr. Roarty confirmed the correctness of the analysis and, as expected, the results of that analysis showed similar analytical results with the new heavier RRVH/HAUP package (Ref. 2). Based upon continued discussion and dialog between Westinghouse and Nuclear Management Company, Westinghouse contracted with Dr. William LaPay to perform an additional review of the 1982 analysis that would consider if inelastic structural behavior would change the conclusions reached in 1982.

Attached is Dr. Lapay's white paper response to that request. This has been reviewed by the undersigned and we agree with the technical judgments expressed by Dr. LaPay. Dr. LaPay's resume is also attached for your reference.

The purpose of Revision 2 was to re-classify this document as non-proprietary. No changes to the content were made. Revision 3 corrected the reference letter number in the attachment.

David H. Roarty[1]
Reactor Component Design & Analysis

Attachments

[1] Official Record Electronically Approved in EDMS 2000.

Assessment of Reactor Vessel Head Drop

Point Beach Unit 1 and Unit 2

May 13, 2005
(reclassified 9/1/2005)

William S. LaPay, PhD

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Purpose

To review the Westinghouse head drop analyses performed in 1982 to confirm the validity of the postulated sequence of events leading to damage to the attached piping which could potentially prevent the removal of decay heat from the core. If this worst case scenario is determined not to be reasonable, then define what is a realistic postulated sequence of events.

References

1. Westinghouse Original Point Beach Nuclear Plant Reactor Vessel Drop Analysis.

   Westinghouse Letter WEP-82-584, PT-SSD-527, "Wisconsin Electric Power Company Point Beach Nuclear Plant Reactor Vessel Head Drop Analysis," November 15, 1982, and Supporting Documentation. (Westinghouse Proprietary)

2. Westinghouse Calculation Note CN-RCDA-05-46, Rev. 2 "Comparison of Original and Replacement Head Drops for Point Beach Unit 1 and Unit 2," approved 4/27/05. (Westinghouse Proprietary)
3. Point Beach Nuclear Plant Drawings, Bechtel Job Number 6118.

   a. Bechtel Drawing C-135, "Containment Structure INT Reinforcing Sections Sh. 2," Rev. 11.
   b. Bechtel Drawing C-325, "Containment Structure Biological Shield Liner Plate Penetrations," Rev. 7.
   c. Bechtel Drawing C-326, "Containment Structure Biological Shield Liner Plate," Rev. 3.
   d. Bechtel Drawing C-2320, "Reactor Steel Supports," Rev. 4.
   e. Bechtel Drawing C-2322, "Location Plans, Major Component Support Structures," Rev. 3.
   f. Bechtel Drawing C-2325, "Containment Structure Biological Shield Liner Plate Penetrations," Rev. 3.
   g. Bechtel Drawing C-2326, "Containment Structure Biological Shield Liner Plate, Rev. 2.

4. Smith, J.O., O.M. Sidebottom, Elementary Mechanics of Deformable Bodies, The Macmillan Company, Toronto, Ontario, 1969.
5. Roark, Raymond J., Formulas for Stress and Strain, McGraw-Hill Book Company, Inc., New York, 3 rd Edition, 1954.
6. Bechtel Specification 0852, Job # 6118, "Structural Design Criteria for the Point Beach Nuclear Plant, Revision 2, 1967.
7. WCAP-14439, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant Units 1 and 2 for the Power Uprate Program," Revision 1, March, 2002.
8. ACI Standard 349, "Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-85)."
9. Gaylord, Edwin H. Charles N. Gaylord, and James E. Stallmeyer, ed., Structural Engineering Handbook, 4th ed., McGraw-Hill, New York, 1997.
10. Manual of Steel Construction, American Institute of Steel Construction, Inc., 7th Edition, 1970.

Summary of Review and Conclusions

A review of the calculations identified in References 1 and 2 was performed. The analyses used accepted energy formulations to define the impact loads. The basic conclusion in Reference 1 is:

> *The results of the analysis for the Point Beach units show that the reactor vessel nozzle stresses are well within allowables. However, the support loads caused by the impact are greater than the critical buckling load of the reactor vessel support columns. These supports cannot be relied upon to absorb enough of the energy of impact to prevent severe damage to the safety injection lines attached to the reactor vessel or to the primary coolant loop piping.*

The 1982 analysis limited itself to elastic behavior of the structures that are impacted. This is conservative. No attempt was made to reflect inelastic material behavior. There is significant energy loss when inelastic behavior is considered. In order to confirm the validity of the conclusion that the impact of the reactor vessel head can cause severe damage so that the core cannot be cooled, a head drop evaluation needs to be performed considering inelastic structural behavior. An assessment was performed to determine if inelastic structural behavior would change the conclusions reached in 1982.

It was concluded from this assessment that the 1982 conclusion is not realistic, inelastic material behavior will absorb significant energy such that there will be no structural failure that would cause loss of coolant to the core. This evaluation has used energy principles to demonstrate that gross plastic failure of the support columns is not likely to occur. The maximum strain levels, for head assembly drop of approximately 45 feet above the flange through air, is 2%, and below 1% for a 20' drop. From this evaluation, it can be concluded that the columns can absorb significant energy and still remain functional. However, if the columns do fail, there is still reserve strength from the primary coolant loop piping that will support the reactor vessel. This is the second scenario that is discussed below.

A second drop scenario is evaluated in which the columns and other potential restraining structures are assumed to fail and only the main coolant pipes are available to restrain the vessel from further drop. This is conservative since this scenario assumes that the reactor vessel safety injection piping fails, columns fail, and the concrete below the vessel ring girder fails, without providing energy absorption. In this evaluation plastic hinges are assumed to form in the loop pipes and the resulting displacement of the reactor vessel assembly is determined. For a 50 foot drop through air, a maximum deformation of less than 7 inches is predicted, after the loop pipes contact the bio-shield wall. For a 15' drop the maximum deformation is less than 3 inches. Assuming that the available flow area is reduced by these deformations, the loss in flow area is under 20%. That is for a 50 foot drop through air, about 80% of the flow area remains, and for a drop of 15 feet the remaining flow area is 95% or the original flow area.

This assessment also confirmed that the concrete supporting the steel structures would not fail considering the maximum loads determined.

Therefore it can be concluded that if a potential head drop was to occur (maximum drop height of 50' considered in analysis), the reactor vessel will be adequately supported either by the supports, or by the primary coolant piping such that there will be no damage to the attached primary coolant piping which would prevent the removal of decay heat from the core. Further, the Reactor Vessel integrity is maintained since it is adequately supported so that it doesn't break and leak out water, except for the safety injection lines that potentially may be lost.

It is recommended that a detailed analysis be performed for the head drop analysis using inelastic nonlinear methods (see Section 5.0). This analysis will provide the permanent documentation supporting the conclusions reached herein that used valid expert opinion and simplified analyses, and provide the documentation that meet NUREG-0612. This evaluation should consider the following:

- Nonlinear time history analyses using different drop heights
- Finite element model that includes gaps (e.g., Δg)
- Actual stress strain curves for the material inelastic behavior for the column support and shear pin, and the hot and cold leg piping.
- The reactor head should impact the vessel with no assumption of inelastic impact considered (i.e., head and vessel are one body after impact).
- The model should also address potential rebound of the vessel and vessel head after impact.

Significance of Inelastic Behavior during Head Drop

An evaluation is performed to determine if there is sufficient energy loss due to inelastic behavior of the impacted structures to prevent the loss of critical components needed to cool the core.

The following assumptions are made that are consistent with Reference 1.

1. A concentric drop of the reactor vessel head onto the reactor vessel is assumed.
2. The vessel head is assumed to be rigid and deflection of the head at impact is neglected.
3. The reactor vessel and vessel head remain together after impact.
4. Seventy-five percent of the available mass of the fuel, water, internals, and remainder of the vessel is helping to dissipate the energy of the drop.
5. The head drop is through air.

Impact Support Scenarios

Two scenarios are evaluated to determine if the impact load can be adequately supported.

Scenario 1: The Reactor Support Columns, Shear Pins, and Concrete Floor supports maintain their structural integrity during impact.

Scenario 2: The Reactor Support Columns are lost during impact along with the safety injection lines. The reactor is supported by the Primary Coolant Loop Piping

Evaluation of Scenario 1

Design Characteristics of Reactor Vessel Vertical Support

Reactor Vessel Vertical Column Design Characteristics per Reference 1:

- Six columns support Reactor
- Columns are 12" Sch 120 pipe
  Area = 36.9 in^2
  Radius of Gyration = r = 4.17 in

- Material A53
  Yield Stress = Fy = 30 ksi (assumes type/grade E/A, F, S/A)
  E = 30E3 ksi
- Column Length = L = 229"
- Pin-Fixed Boundary Condition (K = 0.7)
  KL / r = 38.4 < 40; Region of compressive yield and not buckling
- Capacity based on compressive yield = Pc = 36.9 x 30 = 1107 kips
- Capacity of all six Columns = P_T = 6 x Pc = 6,642 kips

The original load capacity of the columns was reported to be 1157 kips. The difference between Pc and the value of 1157 kips is due to the use of a lower yield stress (30 ksi is used in this evaluation, and 35 ksi was previously used), and it is assumed herein that the column experiences compressive yield failure and not buckling.

Reactor Vessel Column Shear Pin per Reference 1:

- 4" diameter pin
  Area = 12.56 in^2
- Shear Stress = τ = 32 ksi (working stress)
  Faulted Capacity = 1.7 x 32 = 54.4 ksi
- Shear pin support is double shear
  Shear Pin Capacity = Pv = 2 x 12.56 x 54.4 = 1367 kips

Since Pv is larger than Pc, failure of the column will occur before failure of the pin. The shear pin will not see a higher load than Pc since failure of the column will not have a load greater than that defined by a compressive yield failure. Therefore, the shear pin will support the column load Pc, and there is no need to evaluate the shear pin further.

Concrete Supporting Column Support per Reference 6:

- Concrete Compressive Yield Stress = fc' = 4,000 psi

Energy Formulation and Evaluation

From the design characteristics given in Section 4.2.1 it is seen that column will fail before the shear pin, therefore, the column is the critical component in the reactor vessel vertical support structure. The concrete under the support will be evaluated separately.

The column is made of ductile steel, and is assumed to fail in an elastic perfectly plastic behavior as shown below. No credit is taken for strain hardening.

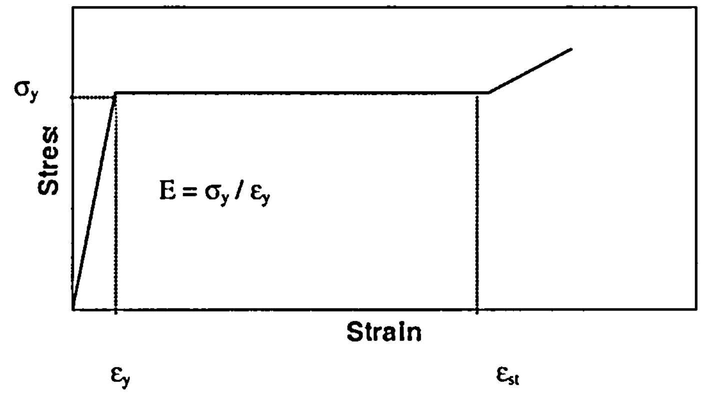

Figure 1– Stress-Strain Curve

The energy loss due to the impact of a mass onto another mass follows References 1 and 2 that uses energy loss formulations given in Reference 5. This formulation is given below for the energy loss factor K.

A moving body of mass M strikes axially one end of a bar of mass M1, the other end of which is fixed. There is a body of mass M2 attached to the struck end of the bar. Then

$$K = \frac{1 + \frac{M1}{3M} + \frac{M2}{M}}{\left(1 + \frac{M1}{2M} + \frac{M2}{M}\right)^2} \qquad (1)$$

Weights are used instead of masses. From Reference 2 the following weights are used:

W = 194,000 lb = weight of replacement head drop weight.

W1 = 137,600 lb = weight of vessel (from flange to nozzles) plus nozzles

W2 = 0.75 x 570,000 lb = 427,000 lb = 0.75 x (weight of vessel plus water in vessel – W1); based on assumption 4 in Section 4.0.

Solving Equation 1, K = 0.272.

The vertical stiffness (Kv) of the vessel, nozzles, and supports is used. Per Reference 2:

Kv = 17.38E3 kip/in per support.

The energy formulation including elastic-perfectly plastic behavior follows Reference 4. This formulation is given below where h is the height of head drop, and Δ is the amount of deformation that includes elastic displacement (Δ_e) and total displacement (Δ) that includes plastic deformation.

$$W\,(K \times h + \Delta) = \tfrac{1}{2} P_T \Delta_e + P_T (\Delta - \Delta_e) \qquad (2)$$

Solving Equation (2) for Δ:

$\Delta = [(K\ h\ W + P_T\ \Delta_c / 2) / (P_T - W)]$ (3)

Where,

$\Delta_c = P_T / Kv = 6{,}642 / 17.38E3 = 0.382$ (4)

Equation (3) is solved for drop heights above the vessel mating surface (El 40' 7 1/8") from 15' to 50'. The results are given in Table 1. Also provided in this table is strain ($\varepsilon = \Delta / L$) in the support column for each drop height. As seen from this table the strain in the column is 2% and under for heights as high as 45', and under 1% for heights below 20'. It is noted that dead weight is not considered in this evaluation, but it will not have significant effect on the results.

Table 1 – Strain in Vertical Support Columns from Head Drop

h feet	Δ inches	Strain ε
50	5.10	0.022
45	4.61	0.020
31.4	3.28	0.014
26.4	2.79	0.012
25	2.65	0.012
20	2.16	0.009
16	1.77	0.008
15	1.67	0.007

Concrete Evaluation

The concrete must support the column load Pc (load based on column yield). Failure of the supporting concrete structure will be local bearing failure under a support column. The supporting floor will be more than adequate to support the total load P_T.

The evaluation follows Reference 8, paragraph 10.15, Bearing Strength.

> Design Bearing Strength on concrete shall not exceed $\phi(0.85\ fc'\ A_1)$, except as follows.
>
> When the supporting surface is wider on all sides than the loaded area, design bearing strength on the loaded area may be multiplied by A_2/A_1, but not more than 2.

After a review of the Reference 3 drawings, it is concluded that the supporting surface is wider on all sides than the loaded area. Therefore, the ratio of A_2/A_1 is taken as 2. The concrete bearing stress (S_B) from the load Pc is:

$S_B = Pc / 2A_1 = 1000 \times 1107 / (2 \times 484) = 1144$ psi

Column base plate is 22"' x 22" (Reference 3d)
$A_1 = 22 \times 22 = 484\ in^2$

Allowable bearing stress = $0.85\ \phi\ fc' = 0.85 \times 0.7 \times 4000 = 2380$ psi
Note, per Reference 8, $\phi = 0.7$ for bearing.

The concrete is acceptable since $S_B < 2380$ psi.

Evaluation of Scenario 2

This scenario is based on the assumption that the columns fail and that only the loop pipes are available to carry the drop load. This scenario also assumes that the direct vessel safety injection piping fails and the concrete below the vessel ring girder fails, without providing energy absorption.

Design Characteristics of Primary Coolant Loop Piping

Hot leg

Per Reference 1:

- Outside Diameter = OD = 34"
- Inside Diameter = ID = 29"
- Thickness = t = 2.5"
- Area = A = $\pi\ (OD^2 - ID^2)/4 = 247.4\ in^2$
- Shear Area = A_v = $247.2 / 2 = 123.7\ in^2$
- Inertia = I = $\pi\ (OD^4 - ID^4)/64 = 30{,}879\ in^4$

Per Reference 7:

- Material A376, Type 316 (ASME Material List Yield Stress = 30 ksi, room temperature)

  Note that per Reference 7 the actual material property is higher than minimum yield stress of 30 ksi.
  Unit 1 yield stress = 32,300 psi
  Unit 2 yield stress = 41,800 psi

  The minimum yield stress value is used.
  The modulus of elasticity at room temperature is 28.3E3 ksi.

Cold leg

Per Reference 1:

- Outside Diameter = OD = 32.25"
- Inside Diameter = ID = 27.5"
- Thickness = t = 2.375"
- Area = $\pi\ (OD^2 - ID^2)/4 = 222.9\ in^2$
- Shear Area = $222.9/ 2 = 111.45\ in^2$
- Inertia = I = $\pi\ (OD^4 - ID^4)/64 = 25{,}026\ in^4$

Per Reference 7:

- Material A376, Type 316 (ASME Material List Yield Stress = 30 ksi, room temperature)

  Note that per Reference 7 the actual material property is higher than minimum yield stress of 30 ksi.

  Unit 1 yield stress = 31,700 psi
  Unit 2 yield stress = 47,000 psi

  The minimum yield stress value is used.
  The modulus of elasticity at room temperature is 28.3E3 ksi.

Concrete Supporting Column Support per Reference 6:

- Concrete Compressive Yield Stress = fc' = 4,000 psi

Energy Formulation and Evaluation

It must be determined if the pipes fail in shear or by the formation of a plastic hinge. It is assumed that the primary coolant loop piping is supported by the shield wall. The piping is assumed to support the reactor vessel as a cantilever beam. The length of the cantilever beam is assumed to be the distance from the nozzle safe-end to approximately half the distance through the biological shield wall.

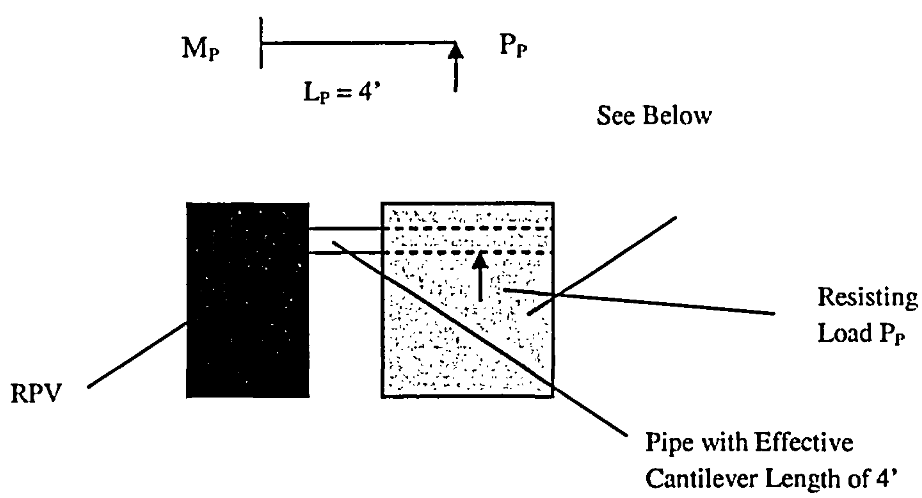

Figure 2 – Piping Modeled as a 4' Cantilever

The plastic moment, when in the flat plateau shown in Figure 1 with a stress equal to σ_y, is defined by Equation (5), per Reference 9, Section 9-1:

$$M_p = \sigma_y Z \qquad (5)$$

where,

M_p = Plastic Bending Moment
σ_y = Yield Stress
Z = Plastic Section Modulus

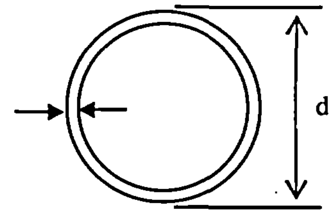

Figure 3 – Piping Cross Section

$$Z = (d^3 / 6) \{1 - [1 - (2t/d)]^3\} \tag{6}$$

Hot Leg

From Equation (6), the plastic section modulus for the hot leg is:

$Z = (34^3 / 6) \{1 - [1 - (2 \times 2.5 / 34)]^3\}$

$Z = 2485.8 \text{ in}^3$

Per Equation (5):

$Mp = 2485.8 \times 30 = 74{,}575$ kip-in

$P_P = M_P / L_P = 74{,}575 / 48 = 1{,}554$ kips

The octahedral shearing stress theory is used to define the shearing yield point.

$\tau_y = \sigma_y / 3^{1/2} = 30 / 3^{1/2} = 17.32$ ksi

$P_v = \tau_y \times A_v = 17.32 \times 123.7 = 2{,}143$ kips

Comparing P_P and P_v it is obvious that the pipe will experience a plastic bending moment before it yields.

Cold Leg

From Equation (6), the plastic section modulus for the cold leg is:

$Z = (32.25^3 / 6) \{1 - [1 - (2 \times 2.375 / 32.25)]^3\}$

$Z = 2124.2 \text{ in}^3$

Per Equation (5):

$Mp = 2124.2 \times 30 = 63{,}726$ kip-in

$P_P = M_P / L_P = 63{,}726 / 48 = 1{,}328$ kips

The octahedral shearing stress theory is used to define the shearing yield point.

$$\tau_y = \sigma_y / 3^{1/2} = 30 / 3^{1/2} = 17.32 \text{ ksi}$$

$P_v = \tau_y \times A_v = 17.32 \times 111.45 = 1{,}930$ kips

Comparing P_P and P_v it is obvious that the pipe will experience a plastic bending moment before it yields.

Energy Formulation

The same energy formulation as used in Section 4.2.2 is used. The pipe is assumed to follow elastic-perfectly plastic behavior with plastic behavior represented by the yield stress. This assumption is not representative of the expected behavior of stainless steel, but is conservative since stainless steel generally does not have a significant flat region of perfectly plastic behavior at its yield stress, but has higher capacity due to strain hardening effect that is not being considered. The elastic displacement (Δ_e) is defined below:

$$\Delta_e = P_P / K' \tag{7}$$

$$K' = 3\,E\,I / L^3 \tag{8}$$

For the Hot Leg:

$$K' = 3 \times 28.3E3 \times 30{,}879 / 48^3 = 23{,}705 \text{ kip/in}$$
$$\Delta_e = 1{,}554 / 23{,}705 = 0.0655"$$

For the Cold Leg:

$$K' = 3 \times 28.3E3 \times 25{,}026 / 48^3 = 19{,}212 \text{ kip/in}$$
$$\Delta_e = 1{,}328 / 19{,}212 = 0.0691"$$

As seen above, the elastic displacement is similar for both the hot leg and cold leg; therefore, the average value is used in this evaluation.

$$\Delta_e = (0.0655 + 0.0691)/2 = 0.0673"$$

The same K value as defined by equation (1) is used, K = 0.272.

There is a 6.5 inch gap, Δg, between the pipe and the shield wall. It is assumed that the total weight of the reactor head assembly, vessel, nozzles, and all contents are assumed to also drop through this gap that adds to the energy of the head drop.

The work done by the plastic hinge is:

$$\text{Work} = M_P\,\theta \tag{9}$$

Assuming small angle deflection, theta is defined as:

$$\theta = \Delta / L \qquad (10)$$

And the relationship between M_P and F_P is:

$$M_P = P_P L$$

Therefore,

$$\text{Work} = M_P \theta = P_P L \Delta / L = P_P \Delta \qquad (11)$$

This is the same relationship as used in Section 4.2.2. Therefore, the formulation as given in Equation (2) may be used modified by the added energy caused by the drop through the gap Δ_g. The value of P_T in Equation (2) is defined as the sum of all of the P_P values associated with the Hot and Cold legs:

$$P_T = 2 \times (1{,}554 + 1{,}328) = 5764 \text{ kips}$$

$$W (K \times h + \Delta) + (W + W1 + W2) \times \Delta_g = ½ P_T \Delta_c + P_T (\Delta - \Delta_c) \qquad (12)$$

Solving Equation (12) for Δ:

$$\Delta = \{[(K h W + P_T \Delta_c / 2) + (W + W1 + W2) \times \Delta_g] / (P_T - W)\} \qquad (13)$$

Equation (13) is solved for drop heights above the vessel mating surface (El 40' 7 1/8") from 15' to 50'. The results are given in Table 2. Considering that the flow area of the pipe is reduced by the deformation, a measure of remaining flow area can be obtained. The formulation using Reference 10, page 6-22, is shown in Figure 4. As seen from this table, at a drop of 50' about 80% of the flow area remains, and for a drop of 15' the remaining flow area is 95% or the original flow area.

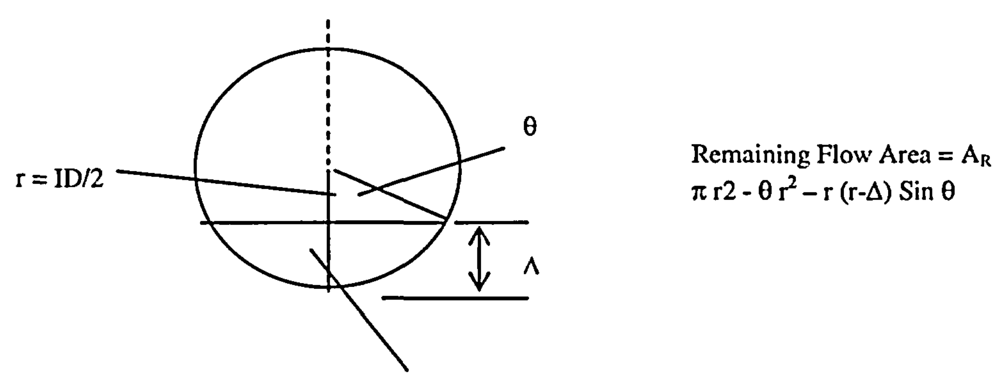

Figure 4 – Remaining Flow Area after Deformation

Table 2 – Remaining Flow Area from Head Drops

h feet	Δ inches	Hot Leg $A_R/\pi\, r2$	Cold Leg $A_R/\pi\, r2$
50	6.77	0.82	0.81
45	6.20	0.84	0.83
31.4	4.65	0.90	0.89
26.4	4.09	0.91	0.91
25	3.93	0.92	0.91
20	3.36	0.94	0.93
16	2.90	0.95	0.94
15	2.79	0.95	0.95

Concrete Evaluation

The Shield Wall concrete must support the pipe loads P_P. Failure of the supporting concrete structure will be local bearing failure under a cold or hot leg pipe at the shield wall interface. The shield wall will be more than adequate to support the total load from the coolant piping.

The same evaluation as done in Section 4.3.1 is used. The evaluation follows Reference 8, paragraph 10.15, Bearing Strength.

The ratio of A_2/A_1 is taken as 2. A_1 is defined as follows using the pipe OD and shield wall dimensions based on Reference 3 (6' for hot leg and 5' for cold leg):

A_1 (Hot Leg) = 34 x 6 x 12 = 2448 in^2

A_1 (Cold Leg) = 32.25 x 5 x 12 = 1935 in^2

The concrete bearing stress (S_B) from the load P_P is:

Hot Leg: $S_B = P_P / 2A_1$ = 1000 x 1,554 / (2 x 2448) = 317 psi

Cold Leg: $S_B = P_P / 2A_1$ = 1000 x 1,328 / (2 x 1935) = 343 psi

Although the actual bearing will likely result in local deformation of a portion of the shield wall concrete, this evaluation supports the contention that the shield wall will not grossly fail.

Allowable bearing stress = 0.85 ϕ fc' = 0.85 x 0.7 x 4000 = 2380 psi
Note, per Reference 8, ϕ = 0.7 for bearing.

The concrete is acceptable since S_B < 2380 psi for both the hot and cold leg piping.

Recommended Further Analyses

A head drop evaluation needs to be performed considering inelastic structural behavior. This analysis will provide the permanent documentation supporting the conclusions reached herein that used valid expert opinion and simplified analyses. This evaluation should consider the following:

- Nonlinear time history analyses using different drop heights
- Finite element model that includes gaps (e.g., Δg)
- Actual stress strain curves for the material inelastic behavior for the column support and shear pin, and the hot and cold leg piping.
- The reactor head should impact the vessel without the assumption of inelastic impact considered (i.e., head and vessel are one body after impact). Impact shall consider more realistic material behavior and dynamic structural response.
- The model should also address potential rebound of the vessel and vessel head after impact.

ENCLOSURE 2

POINT BEACH UNIT 2
REACTOR VESSEL CERTIFIED MATERIAL TEST REPORTS (CMTR)
(NON-PROPRIETARY)

(13 pages follow)

Point Beach Unit 2

Reactor Vessel CMTRs

W.C [illegible]

Form 7

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NATIONAL FOR[illegible] COMPANY

RVG- 0000003619

HOLLOW BORED FORGING SPECIALISTS

IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No. 61001405110

TO The Babcock & Wilcox Company
Boiler Division
Barberton, Ohio 44203

CUSTOMERS ORDER NO. 32920E
N.F. ORDER NO. 79-3847
DATE Sept. 18, 1968

RVG 000003619

NATIONAL FORGE COMPANY

HOLLOW BORED FORGING
SPECIALISTS

IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No. 61001405110

TO The Babcock & Wilcox Company
Boiler Division
Barberton, Ohio 44203

CUSTOMERS ORDER NO. 32920B
N.F. ORDER NO. 79-3847
DATE Sept. 18, 1968

Table is ... head ... information

RVG-0000003619

Purchaser: Babcock & Wilcox Co.
Contract No. 610-014-05110

No. NEMH-42-4 Rev. 2
Date March 15, 1968

T.

RVG-0000003617

NATIONAL FORGE COMPANY
HOLLOW BORED FORGING SPECIALISTS

IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No. 61001405110

TO The Babcock & Wilcox Company
Boiler Division
Barberton, Ohio 44203

CUSTOMERS ORDER NO. 329208
N. F. ORDER NO. 79-3847
DATE Sept. 18, 1968

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NATIONAL FORGE COMPANY

HOLLOW BORED FORGING
SPECIALISTS

IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No.
61001405110

TO The Babcock & Wilcox Company
Boiler Division
Barberton, Ohio 44203

CUSTOMERS ORDER NO. 32920
N.F. ORDER NO. 79-3847
DATE Sept. 18, 1968

CORRECTED COPY   RVG- 0000003617

National Forge Co. -79-3847-02-#002

ASSUMED TO BE 42-4
REE
10/29/93

Customer Babcock & Wilcox Co.   PO # 32920   No. 42-2   Rev. 2

Contract No. 610-014-05110   Date March 15, 1968

Westinghouse Non-Proprietary Class 3

RVG-0000003618

B10-V

NATIONAL FORGE COMPANY
HOLLOW BORED FORGING
SPECIALISTS
IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No.
61001405110

TO Babcock & Wilcox Co.
Boiler Div.
Barberton, Ohio 44203

CUSTOMERS ORDER NO. 329202
N.F. ORDER NO. 79-3847
DATE Nov. 28, 1968

RVG-0000003618

NATIONAL FORGE COMPANY

HOLLOW BORED FORGING
SPECIALISTS
IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No.
61001405110

To Babcock & Wilcox Co.
Boiler Div.
Barberton, Ohio 44203

CUSTOMERS ORDER NO. 32920E
N.F. ORDER NO. 79-3847
DATE Nov. 28, 1968

CORRECTED COPY

RVG- C000003618

National Forge Co. -79-3047-02-500

[illegible]

Purchaser Babcock & Wilcox Co.

Contract No. 610-014-05110

PO # 32920

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RBE
10/29/93

~~NEW~~ 42-2 Rev. 2

Date ~~March~~ 15, 1968

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Form 301 R

NATIONAL FORGE COMPANY
HOLLOW BORED FORGING
SPECIALISTS
IRVINE, WARREN COUNTY, PENNSYLVANIA

TO The Babcock & Wilcox Company
Barberton, Ohio

CUSTOMERS ORDER NO. 61001405110 32920E
N.F. ORDER NO. ~~79-~~3847
DATE March 16, 1970

KVG- 0000003620

Form 205

NATIONAL FORGE COMPANY
HOLLOW BORED FORGING SPECIALISTS

IRVINE, WARREN COUNTY, PENNSYLVANIA

Contract No. 61001405110

TO The Babcock & Wilcox Company
Barberton, Ohio

CUSTOMERS ORDER NO. 329203

N.F. ORDER NO. 79-3847

DATE March 16, 1970

Table for heat treating information SYG- 000-0003620

Purchaser Babcock & Wilcox Co.
Contract No. 610-014-05110

No. NEMH-42-4 Rev. 2
Date March 15, 1968