NMC
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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

June 9, 2005

NRC 2005-0072
10 CFR 50.90

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

Point Beach Nuclear Plant Unit 2
Docket 50-301
License No. DPR 27

Supplement 3 to Request for Exigent Review of Heavy Load Analysis and Response to Request for Additional Information

References:
1. NMC Letter to NRC Dated April 29, 2005
2. NMC Letter to NRC Dated May 13, 2005
3. NMC Letter to NRC Dated May 19, 2005
4. NMC Letter to NRC Dated June 1, 2005
5. NMC Letter to NRC Dated June 4, 2005

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

References 2 and 3 submitted supplements to the proposed amendment to provide the results of additional assessments and to incorporate additional technical justification for the proposed amendment. Additionally, Reference 2 retracted the proposed amendment for PBNP Unit 1 and proposed to apply the reactor vessel head (RVH) lift assessment on a one-time basis for the upcoming lift of the Unit 2 RVH. References 4 and 5 provided responses to questions posed by Nuclear Regulatory Commission (NRC) staff.

During telephone conferences between NMC personnel and NRC staff on June 6 and 8, 2005, the staff requested additional information in support of the proposed amendment. The NMC response to the staff's questions is provided in Enclosure 1. Enclosure 4 provides plant drawings requested by the staff, depicting dimensional clearances in the region surrounding the reactor vessel. NMC is also submitting results of finite element analyses (FEAs) of the postulated RVH drop scenario prepared by Sargent & Lundy and also by Automated Engineering Services

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Corporation. Enclosure 2 submits Sargent & Lundy Calculation, "Analysis of Postulated Reactor Head Load Drop onto the Reactor Vessel Flange", dated June 8, 2005. Enclosure 3 submits Automated Engineering Services Corporation Calculation, "Finite Element Analysis of Reactor Vessel Head Drop on Reactor Vessel Flange", dated June 8, 2005.

These two detailed and independently performed analyses have been reviewed and accepted by NMC. The Sargent & Lundy analysis followed the same methodology as used for the Prairie Island Nuclear Generating Plant FEA. The Sargent & Lundy and Automated Engineering Services analyses are consistent and each demonstrate that reactor vessel deflection following a postulated RVH drop would be less than four inches. This amount of deflection would be insufficient to cause a loss of decay heat removal capability.

The original 1982 analysis conducted by Westinghouse reached conclusions that the effects of a RVH drop would cause damage to the reactor vessel supports and reactor coolant system (RCS) piping upon impact, and postulated (by engineering judgment) that a potential to lose decay heat removal capabilities existed.

While assessing various methods to administratively control the final effects of such a postulated event via multiple mitigation strategies, NMC continued to refine the analytical understanding of the postulated RVH drop.

The following analytical activities were performed.

1. An evaluation, using the same methodology as the 1982 analysis, was conducted by Westinghouse. This evaluation provided additional rigor and continued where the 1982 analysis concluded. This evaluation was submitted to the NRC in Reference 2.

2. FEAs were conducted by two independent vendors to further characterize the effects of the RVH drop. Each vendor utilized slightly different modeling approaches and FEA analytical software, accounting for minor differences in actual results obtained.

Although the impact from a postulated RVH drop would cause reactor vessel deflection, these analyses demonstrate that the 6.5 inch gap that exists between the RCS piping and the shield wall, and the 4 inch gap that exists between the core deluge line and the shield wall, are adequate to preclude damage that would result in a loss of decay heat removal. Based on the results from these multiple analytical approaches, NMC has concluded that adequate reactor core cooling capability would be maintained following the expected deflection of the reactor vessel from a postulated RVH drop. The details of these analyses are provided in the enclosures.

Also provided in Enclosure 5 are a Westinghouse authorization letter, accompanying affidavit, Proprietary Information Notice and Copyright Notice for specified pages (marked "Proprietary") of the calculations contained in Enclosures 2 and 3.

Since the document pages listed above as Proprietary contain information proprietary to Westinghouse Electric Company, they are 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-05-2004) and be addressed to J. A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

NMC has determined that this additional information for the proposed amendment, including deletion of the temporary modification, remains bounded by the No Significant Hazards Consideration Determination submitted May 19, 2005 (Reference 3).

This letter contains one change to a previous commitment. Since the analyses conclude that adequate core cooling will continue to be maintained with normally installed plant equipment, NMC will not implement the temporary core cooling modification discussed in Reference 2.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 9, 2005.

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

Enclosures (5)

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

bcc: (with enclosures)
File

bcc: (w/o enclosures)

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ENCLOSURE 1

SUPPLEMENT 2 TO REQUEST FOR REVIEW OF HEAVY LOAD ANALYSIS

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

In letters to the Nuclear Regulatory Commission (NRC) dated May 13 and 19, 2005, Nuclear Management Company (NMC) proposed an amendment to the license for Point Beach Nuclear Plant (PBNP) Unit 2, to support a change to the PBNP Unit 2 licensing basis regarding control of heavy loads.

The NRC staff requested additional information (RAI) in support of their review of the proposed amendment. This included a request for revised bounding conditions for the postulated reactor vessel head (RVH) drop event based on the finite element analyses (FEAs) being submitted in Enclosures 2 and 3. Requested information included an assessment of the postulated RVH drop sequence; an analysis of the postulated reactor vessel (RV) displacement and its impact on reactor coolant system (RCS) piping, bottom mounted instrumentation (BMI) tubes, fuel integrity, and radiological consequences; and RCS makeup capability.

The staff also requested plant drawings to allow the staff to verify dimensional clearances around the RV, its associated supports, and RCS piping in the vicinity of the RV. The requested drawings are provided in Enclosure 4.

To facilitate the staff's review, the following description of references to previous submittals that remain germane to the proposed amendment are provided.

References:

1. NMC Letter to NRC Dated April 29, 2005 (NRC 2005-0055)
   Original amendment request;
   Superceded in its entirety by Reference 2; no longer germane.

2. NMC Letter to NRC Dated May 13, 2005 (NRC 2005-0063)
   Supplement 1;
   This supplement replaced Reference 1 in its entirety.

3. NMC Letter to NRC Dated May 19, 2005 (NRC 2005-0064)
   Supplement 2 and Response to Request for Additional Information;
   Provided the NMC response to the staff's questions;
   Provided a revised No Significant Hazards Consideration Determination.

4. NMC Letter to NRC Dated June 1, 2005 (NRC 2005-0067)
   Response to Request for Additional Information;
   Provided response to radiological dose and BMI questions that are revised in this supplement; also provided drawings and a BMI tube analysis;
   The responses to questions in this letter are no longer germane.

5. NMC Letter to NRC Dated June 4, 2005 (NRC 2005-0070)
   Response to Request for Additional Information;
   Provided response to probabilistic risk assessment questions.

The following information is provided in response to the staff's request and to delineate the final analysis results for the proposed amendment. The information is arranged by topic and provided under the titled sections below.

Postulated RVH Drop Sequence

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

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

The 1982 analysis referred to in Wisconsin Electric Power Company letter dated November 22, 1982, is contained in Westinghouse letter WEP-82-584, "Reactor Vessel Head Drop Analysis," dated November 15, 1982. Westinghouse performed the evaluation of the effects of a postulated reactor vessel head drop accident as described in NUREG-0612 for the Point Beach Nuclear Plant. A head drop is postulated to occur during refueling when the head is manipulated above the reactor vessel. The polar crane is postulated to fail and the head is assumed to fall concentrically onto the reactor vessel.

Administrative controls have been established to limit the maximum RVH drop height to 26.4 ft. This drop height has been utilized in the FEAs discussed below.

Analysis of Reactor Vessel Deflection

Based on NMC's assessment of the FEAs provided in Enclosures 2 and 3, the following bounding conditions apply.

Following the postulated RVH drop, the RV deflects less than four inches. The RCS piping and safety injection (SI) core deluge piping does not fail as a result of this RV deflection.

Effect of RV Deflection on RCS and Safety Injection Piping

The impact of the postulated RV deflection on the attached RCS piping was assessed. This assessment was performed by the NSSS vendor and based on the following parameters.

Simple finite element models of the hot leg and cold leg, including non-liner material properties, were built. Inputs included a maximum 4 inch vertical downward displacement at the reactor vessel nozzle supports, and no existing vertical structural constraints of the RCS piping between the reactor vessel and the steam generator (for hot leg) or reactor coolant pump (for cold leg). The load was assumed to be applied as a vertical static displacement at the reactor vessel nozzle supports. Equipment constraint points were conservatively assumed to be fixed to predict maximum stress.

The analysis concluded that the maximum stress intensity is about 0.5 S_u, which is less than the ASME Section III Code, Appendix F limit of 0.9 S_u. The maximum stress and strain occurs near the reactor vessel inlet nozzle at the bottom of the pipe. The maximum strain obtained was about 17.4%. Typical strain values at the ultimate tensile strength for Type 304 stainless steel are greater than 40%. Therefore, the maximum strain as a percent of the estimated limit is about 0.44 (17.4/40).

The assessment demonstrates, with margin, that the reactor coolant pipe can be subjected to the postulated displacement and maintain maximum stress and strain values that are reasonable with respect to applicable ASME Code faulted limits.

Additionally, the integrity of the two 6 inch safety injection 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 SI lines are more flexible than the RCS piping, leads to the conclusion that the integrity of the SI lines is bounded by the assessment for the RCS piping.

Consequently, the temporary core cooling modification discussed in NMC letters dated May 13 and 19, 2005 (References 2 and 3), is not required as a cooling water source. Therefore, this temporary modification (TM 2005-008) will not be implemented.

Effect of RV Deflection on Bottom Mounted Instrument Tubes

The RV bottom mounted instrument (BMI) tubes were analyzed to remain intact following a postulated RV deflection of 23 inches (analysis provided in Reference 4). As

discussed in the June 1, 2005, NMC response to NRC questions 6, 7, 8 and 9 (Reference 4), an informal computer analysis was performed that restrained the BMI tubes axially at the first spring restraint. The allowed downward deflection of the reactor vessel for this condition was approximately 6 ½ inches of travel to maintain stresses below the 3.0 S_c limit. This provides an upper bound for the effects of friction at the supports.

Since the calculated RV deflection is less than 4 inches, the above analysis remains bounding. For conservatism, a scenario that postulated failure of all 36 BMI tubes was also assessed. This assessment determined that the resulting RCS leakage due to gravity drainage of RCS water through the failed tubes is well within the makeup capability of a single residual heat removal (RHR) or SI pump. Therefore, RCS inventory and adequate core cooling would be maintained.

In the June 1, 2005, submittal, NMC provided information regarding the BMI tubes in response to NRC questions 6, 7, 8 and 9. For ease of reference, those questions are restated below in their entirety with the updated NMC responses following.

NRC Question 6 (Reference 4):

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

NMC Updated Response:

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

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

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

The maximum stress for a 23 inch downward displacement of the RPV is 55,385 psi at node '30 F'. This is below the 56,400 psi allowable of 3.0 S_c.

NRC Question 7 (Reference 4):

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

NMC Updated Response:

The calculation utilizes an 'AUTOPIPE' pipe model to establish the maximum vessel displacement required to over stress the BMI tubes. The model uses a thermal growth methodology, taking into account the gaps at the supports. This calculation was developed to support the conclusion reached in the May 13, 2005, letter (Reference 2), which states that based on engineering judgment, the BMI connections remain intact. The clearance between the bottom of the first support, which is a spring can support, and the floor is 1 inch. The analysis conservatively assumes that the entire run is only allowed to deflect 1 inch downward before floor contact is made.

NRC Question 8 (Reference 4):

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

NMC Updated Response:

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

NRC Question 9 (Reference 4):

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

NMC Updated Response:

The calculation associated with the BMI analysis (Enclosure 4 of Reference 4) used a stress intensification factor (SIF) of 1.3 to account for concerns with stresses on welds,

at the connection to the reactor vessel, and at the coupling welds located along the BMI tubes. This SIF is in accordance with ANSI B31.1-1967 to account for the higher stresses at fillet welded joints.

RCS Makeup Capability

Although damage to the BMI tubes is not anticipated, there are numerous flow paths available to provide makeup to the reactor vessel. The various flow paths are listed in the table below. Available pumps include:

- Two Residual Heat Removal pumps each with a design capacity of 1560 gpm at a design head of 280 ft.
- One Safety Injection pump with a design capacity of 700 gpm at a design head of 2600 ft.
- Two Charging pumps with a capacity of 60 gpm each.

	System	RV inlet
1	Safety Injection Train A via SI-878D	A Cold Leg
2	Safety Injection Train B via SI 878B	B Cold Leg
3	Safety Injection Train A via SI-878A	B Core Deluge
4	Safety Injection Train B via SI-878C	A Core Deluge
5	RHR via 2RH-720	B Cold Leg
6	RHR via 2SI-852A	A Core Deluge
7	RHR via 2SI-852B	B Core Deluge
8	Charging via 2CV-1298	A Cold Leg
9	Charging via 2CV-1296	B Cold Leg
10	Charging via Auxiliary Spray Line	B Hot Leg

Therefore, RCS inventory and adequate core cooling would be maintained.

Effect of RVH Drop on Fuel Integrity and Radiological Consequences

In the May 13, 2005 letter, NMC provided the following dose assessment for the RVH drop event.

> For plants that currently have Westinghouse-supplied drive shafts and Westinghouse-supplied fuel, Westinghouse has concluded that in the event of a reactor vessel head drop, skeletal damage to the 14x14 fuel assembly structure could occur, but that the fuel cladding integrity would be maintained (Westinghouse Nuclear Safety Advisory Letter 04-6 and WCAP 9198, Revision 1, "Reactor Vessel Head Drop Analysis"). This conclusion applies to PBNP Unit 2, because it has Westinghouse supplied drive shafts and the Westinghouse 14x14

VANTAGE 422+ fuel. In addition, at the end of the operation cycle for Unit 2, no fuel defects were present. The core reload for Unit 2 Cycle 28 replaced 36 thrice burned assemblies with fresh 422V+ fuel assemblies. Twenty of the fresh fuel assemblies have rod control cluster assemblies (RCCAs); the remaining 13 RCCA locations contain once-burned fuel. With this core configuration, the load placed on the drive shafts of the RCCAs will primarily be transferred to fuel assemblies with fuel rods that have never been irradiated. Since there is no loss of clad integrity due to the postulated event, there is no release path introduced for the fission-product gases contained in the fuel cladding gap to escape.

The staff subsequently noted that they had not previously approved WCAP-9198. Instead, the staff requested that dose calculations for this event be performed using a clearly bounding source term assumption, such as 100% clad gap release for all previously burned assemblies, and the initial conditions described in Enclosure 2 of Reference 2.

In lieu of performing a specific evaluation of the impact on the fuel from a RVH drop at Point Beach, NMC performed the following bounding analysis, which was communicated in Reference 3.

An evaluation was performed to assess the offsite and control room dose consequence following a Unit 2 RVH drop. The drop scenario assumes that the event occurs when the RVH is concentric to the vessel. A RVH drop can be postulated as an initiator of a loss of coolant accident (LOCA) under shutdown conditions. Because the event circumstances are similar to a fuel handling accident (FHA), the dose evaluation is based upon a combination of LOCA and FHA input assumptions as they apply to the event scenario. For purposes of providing a bounding source term, the postulated 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:

- Unit 2 has been shut down for greater than 30 days;
- Unit 2 Cycle 28 reload core in place (36 fresh fuel assemblies; 85 previously burned assemblies);
- Containment equipment hatch and personnel airlocks are closed (equipment hatch on and bolted, one access door closed in each airlock, interlocks operable);
- Purge supply/exhaust system isolation valves shut;
- Containment closure has been established in accordance with checklist CL-1E, which is the procedure that establishes and maintains containment penetration status;

This bounding evaluation remains applicable to this license amendment request and conservatively bounds the reactor vessel and RCS damage predicted by the FEAs performed and submitted as part of this correspondence.

Per the recent communications with the NRC, the impact on the dose consequence post-RHV drop of incorporating current regulatory guidance on assumptions related to emergency core coolant system (ECCS) leakage and unfiltered inleakage is to be affirmatively addressed. We have revised our June 1, 2005, submittal information regarding the radiological consequences in response to NRC questions 10 and 11 to address staff comments concerning control room inleakage and ECCS leakage. For ease of reference, those questions are restated below in their entirety with the updated NMC responses following.

NRC Question 10 (Reference 4):

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

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

NMC Updated Response:

The reason the ECCS leakage rate multiplier was not originally applied in the RVH drop dose evaluation provided in NMC letter dated May 19, 2005, was based on information contained in various regulatory guidance documents, in addition to knowledge of past leakage measurements. The original dose evaluation considered the guidance of Regulatory Issue Summary (RIS) 2001-0019 and Position 1.3.2, "Reanalysis Guidance", of Regulatory Guide (RG) 1.195, as well as other sections of this RG. RIS 2001-0019, Position 1, states that an assumption made in a licensee's analysis supporting a docketed amendment request to be part of the current design basis if the staff relied upon that assumption when evaluating whether the NRC requirements were met in granting the licensing amendment. Therefore, since the RVH drop for PBNP was considered as an accident initiator of a LOCA, the CLB LOCA radiological accident analysis was reviewed to determine the portions of the analysis that would be impacted by this event. The CLB LOCA radiological accident analysis for the control room does

not include the factor of two multiplier on ECCS. The CLB LOCA radiological accident analysis was incorporated into the FSAR under NRC safety evaluation (SE) dated July 9, 1997. All other design basis radiological accident analyses were approved under NRC SE dated July 1, 1997.

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

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

The intent of the dose assessment evaluation provided was to provide reasonable assurance via a scoping analysis that for the postulated Unit 2 reactor vessel head drop over the Cycle 28 reload core at 30-days post-shutdown, the dose consequences off-site and to the control room are clearly bounded by the current licensing basis LOCA radiological accident analysis and do not create a more severe hazard.

Rather than providing further technical information to justify the presumption that the ECCS leakage rate is limited to 400 cc/min post-accident, NMC is adhering to current guidance and doubling the administrative leakage rate to 800 cc/min post-accident for this assessment. The impact of adhering to the regulatory guidance assumption for design basis radiological consequences for ECCS leakage post-LOCA (RG 1.195) results in a doubling of the previously calculated control room operator dose since the rate of ECCS leakage is directly proportional to the dose. The estimated control room operator thyroid and whole body doses provided in NMC letter dated May 19, 2005, are 1.4 rem and 2.05E-03 rem, respectively. Applying the factor of two multiplier on the

ECCS leakage rate result in the estimated control room operator thyroid and whole body doses post-RVH drop of 2.8 rem and 4.1E-03 rem, respectively. The offsite doses remain unchanged from NMC letter dated May 19, 2005, since these values are scaled on LOCA dose consequences that already include the doubled ECCS leakage rate.

The current licensing basis radiological consequence of a design basis LOCA bounds the dose consequence of the postulated Unit 2 RVH drop, and the LOCA remains the limiting event for control room habitability.

NRC Question 11 (Reference 4):

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

NMC Updated Response:

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

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

previously evaluated. Incorporation of the measured unfiltered inleakage does not change this conclusion since it is based on a scaled LOCA dose, which is the design basis radiological analysis for control room habitability for PBNP. However, with the understanding that staff is requesting a technical justification for implicitly not accounting for the measured unfiltered inleakage in the RVH drop dose evaluation, NMC will incorporate the impact of the measured unfiltered inleakage on control room operator dose post-RVH drop.

In order to assess the impact of the measured unfiltered inleakage 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. When the control room emergency filtration system (CREFS) is in operation (mode 4 of the control room ventilation system), it is assumed that 4950 cfm of outside air is supplied to the control room to provide filtered air which will pressurize the control room. Since the efficiency of the CREFS filters is 95% for elemental/organic and 99% for particulate, the activity is entering the control room via the ventilation system at an estimated rate of 250 cfm for elemental/organic and 50 cfm for particulate. The current licensing basis CRH analysis assumes that while in mode 4, the unfiltered inleakage is 10 cfm. The CLB ECCS leakage pathway is assumed to contain only elemental iodine. Therefore, under the current licensing basis analysis, the dose to the operator via the ECCS leakage path is primarily due to the elemental iodine activity delivered through the ventilation system.

Recent tests of unfiltered inleakage to the control room while in mode 4 determined that unfiltered inleakage is approximately 100 cfm. This information was previously discussed in NMC letters to the NRC dated December 5, 2003 and September 29, 2004. 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 dose increases by a factor of 1.35 (350 cfm / 260 cfm). Since the ECCS leakage pathway is assumed to contain only elemental activity, the control room doses increase at most by a factor of 1.35.

The cumulative impact on the RVH drop dose consequence of doubling the administrative limit of the ECCS leakage rate (factor of 2) and incorporating the measured unfiltered inleakage (factor of 1.35) is an increase proportional to a factor of 2.7 (i.e., 2 x 1.35). Using the estimated control room operator doses provided in NMC letter dated May 19, 2005, the resulting control room thyroid and whole body doses become 3.8 rem and 5.5E-03 rem, respectively.

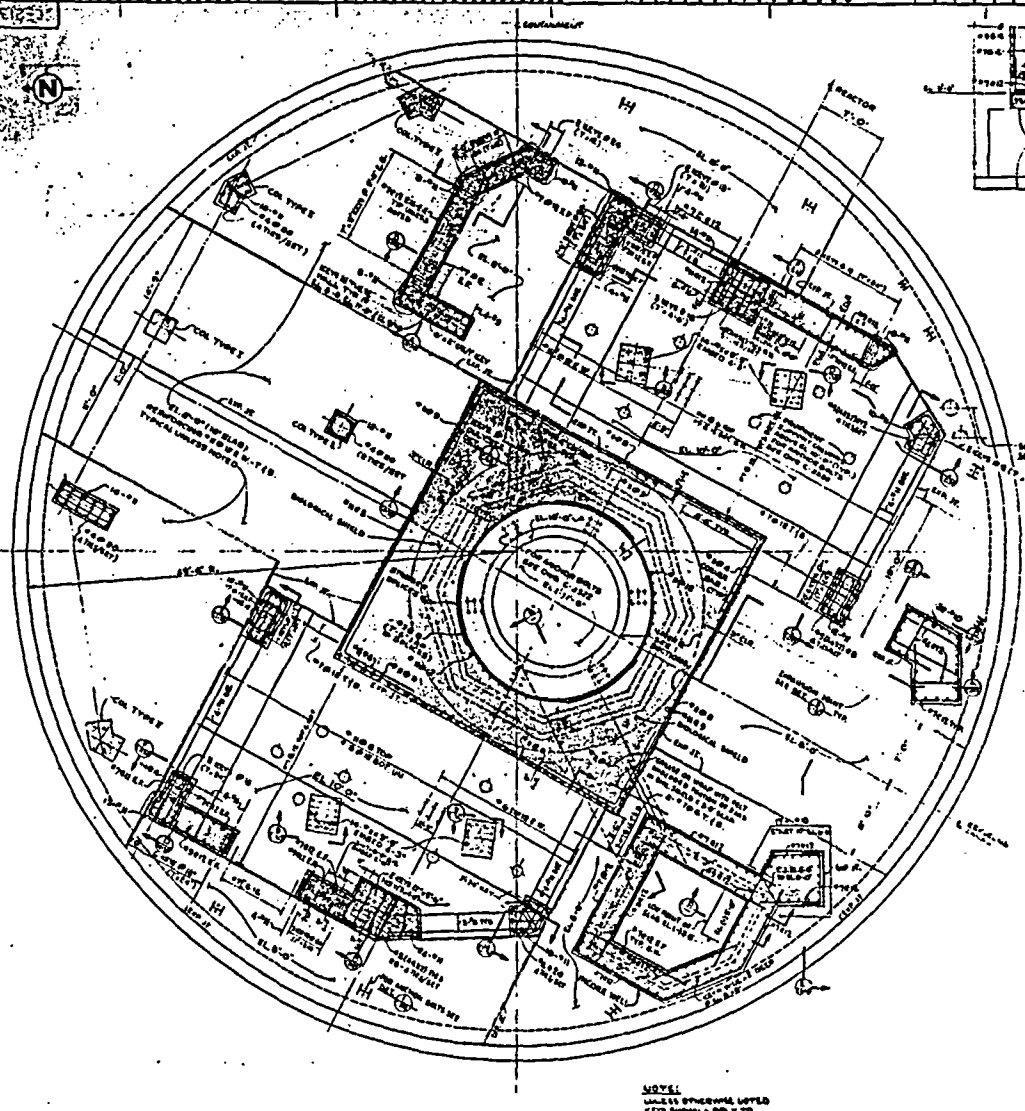
The current licensing basis radiological consequence of a design basis LOCA bounds the dose consequence of the postulated Unit 2 RVH drop, and the LOCA remains the limiting event for control room habitability.

The source term assumed for the Unit 2 Cycle 28 RVH drop event provides a significant margin of safety. The projected control room thyroid and whole body doses of 3.8 rem and 5.5E-03 rem, respectively are well within GDC 19 limits.

ENCLOSURE 4

ENGINEERING DRAWINGS
PBNP REACTOR VESSEL EXTERIOR REGION DIMENSIONAL CLEARANCES

(12 pages follow)

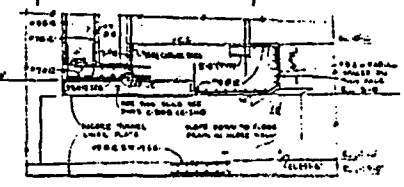


PLAN EL 10-0"

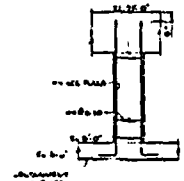
NOTES

1. FOR NOTES & REFERENCE DRAWINGS SEE DWG C-2128
2. FOR CONCRETE MATERIAL TEST CHANNELS SEE DWG C-2128 FOR LOCATION OF ELEVATION
3. THE EXPANSION JOINT SEALANT WAS REPLACED WITH BOND COMPOUND FOR JOINT SEALANT DURING USING PRE AND BRIDGE. THERE WERE FOUR 6-INCH DIAMETER CORE HOLES DRILLED IN THE CONCRETE FLOOR TO INSPECT THE LINER PLATE. REVIEW AND SEE DR FOR ADDITIONAL INFORMATION.

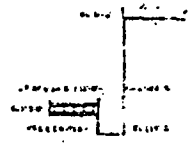
NOTES:
UNLESS OTHERWISE NOTED
NOTE SHOWN APPLY TO
CONSTRUCTION JOINTS AT
EL. 10'-0" & EL. 17'-0"
FOR KEY DETS SEE DET



SECTION



TYPICAL COLUMN ELEVATION



SECTION

BECHTEL
SAN FRANCISCO

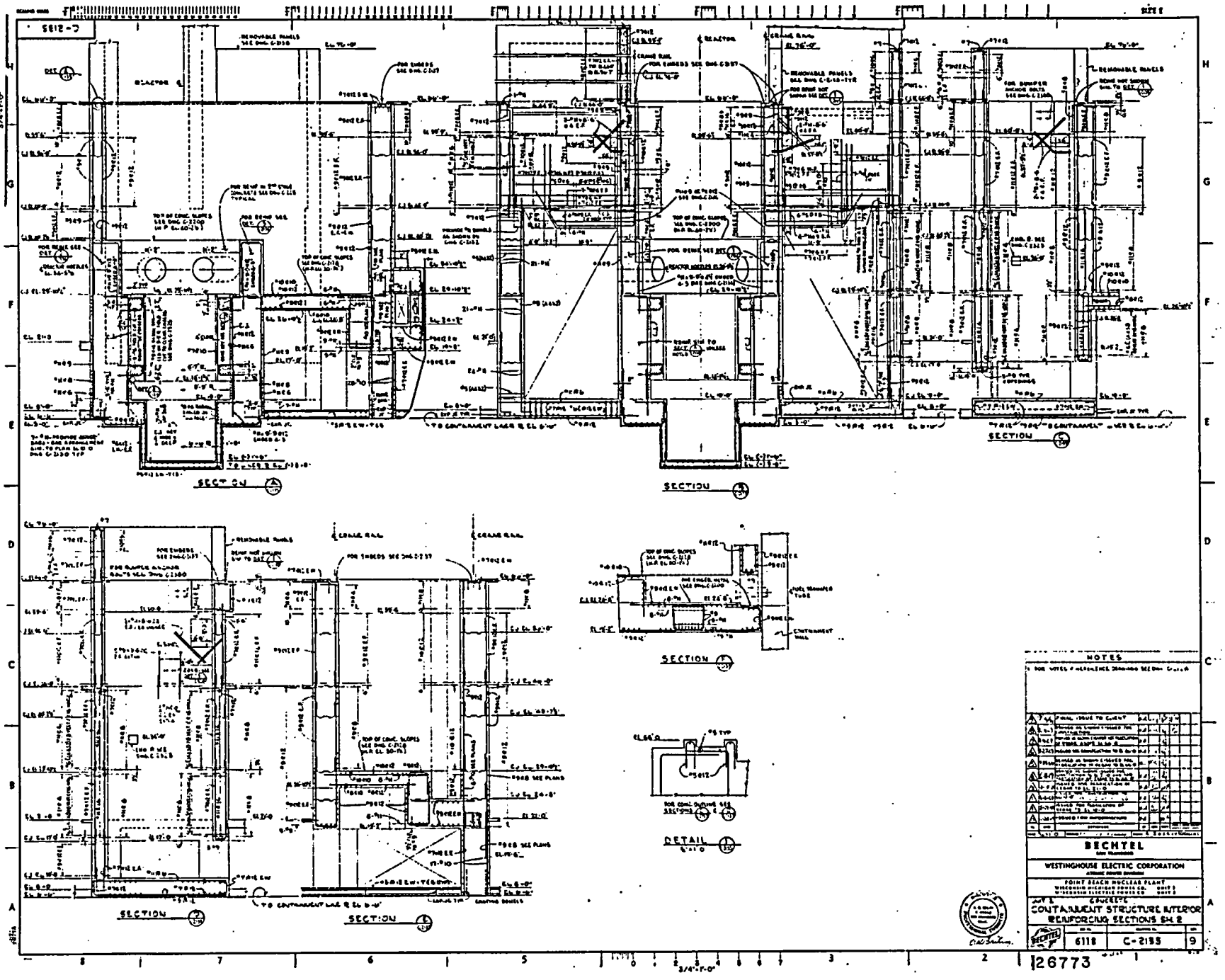
WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIVISION

POINT BEACH NUCLEAR PLANT
WISCONSIN MICHIGAN POWER CO UNIT 1
WISCONSIN ELECTRIC POWER CO UNIT 2

UNIT 2 CONCRETE
CONTAINMENT STRUCTURE INTERIOR
REINFORCING PLAN EL 10-0" & DETAILS

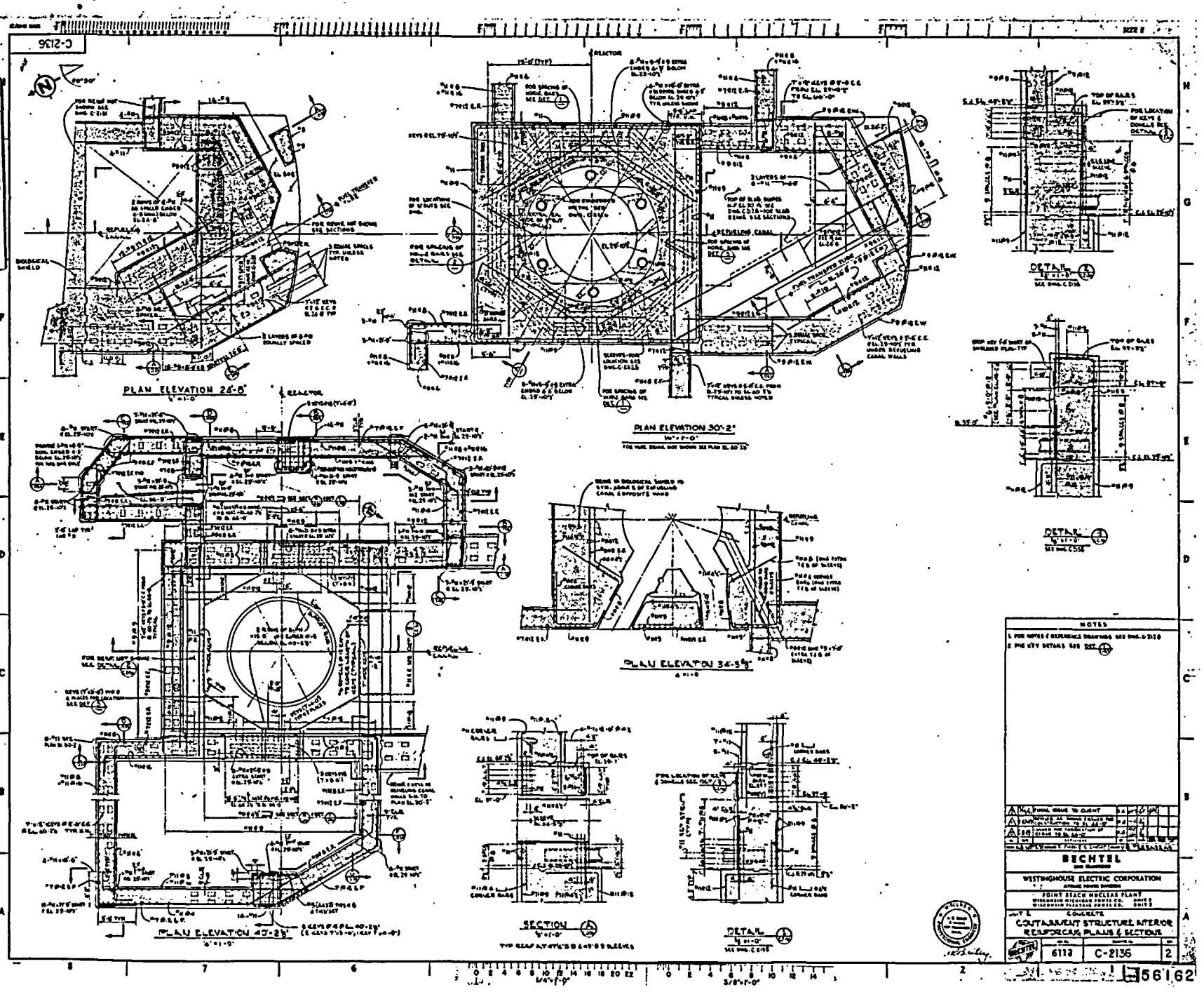
6118	C-2130	5

56159 | PB 025020002004 05



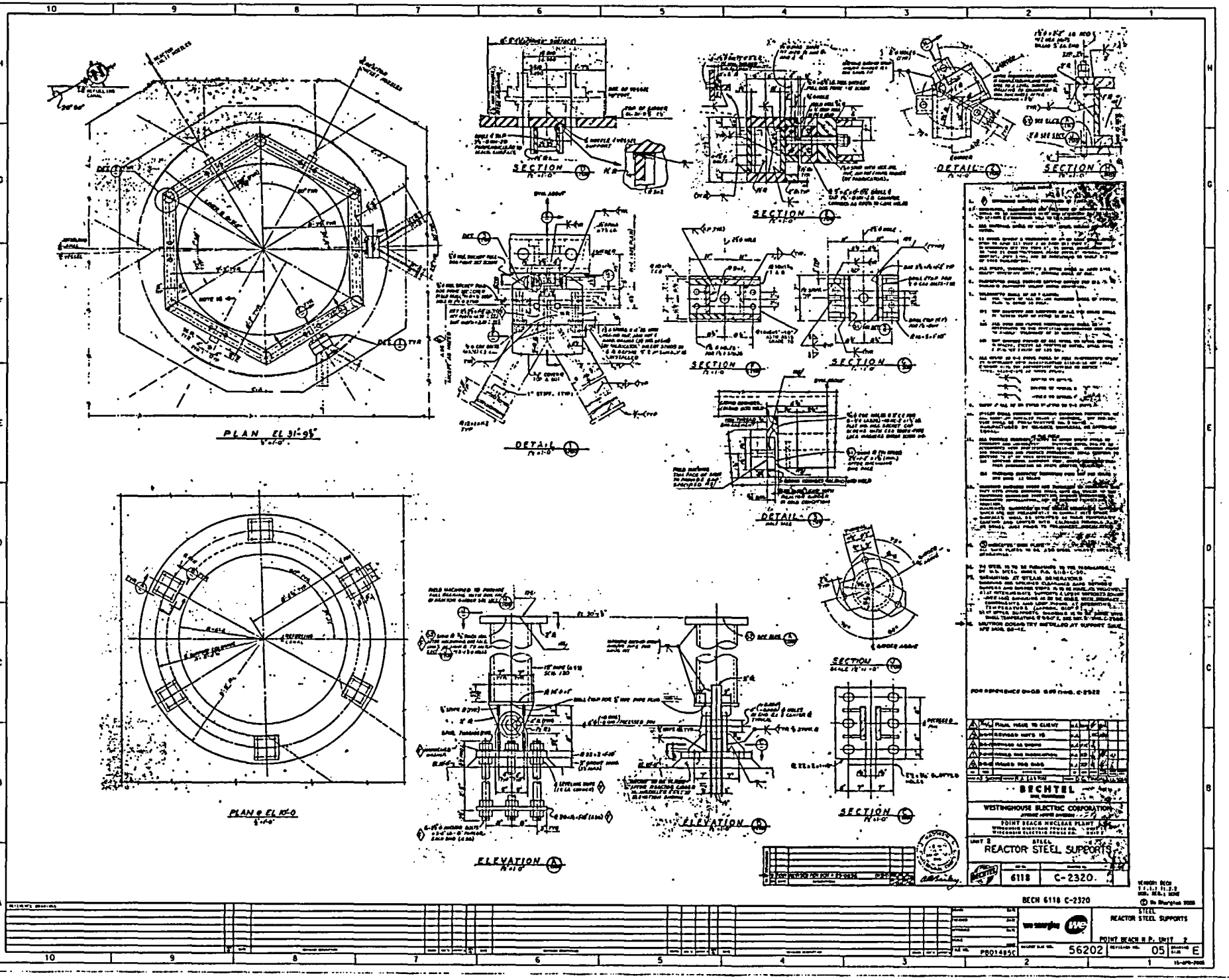
30XJAN 1975

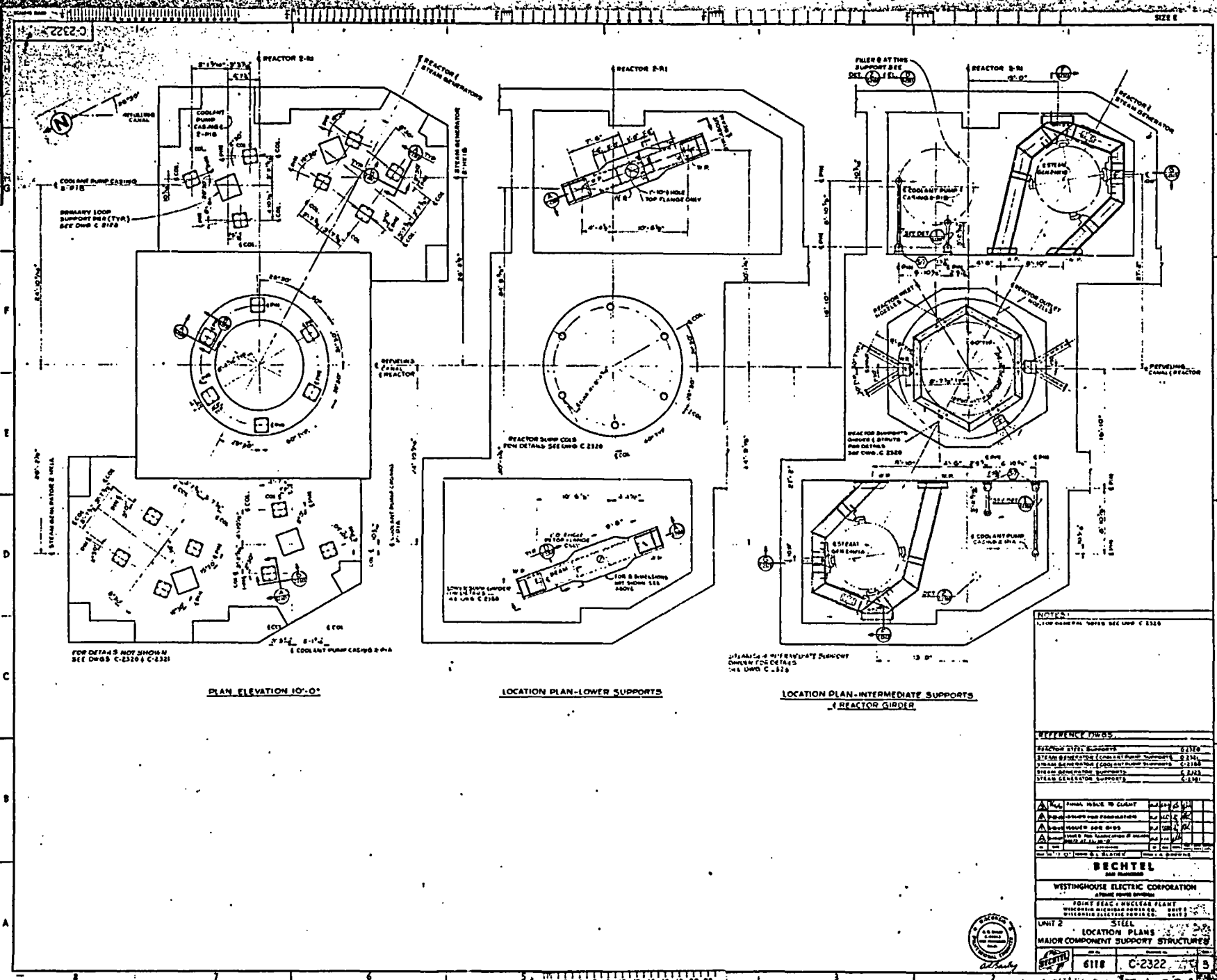
126773

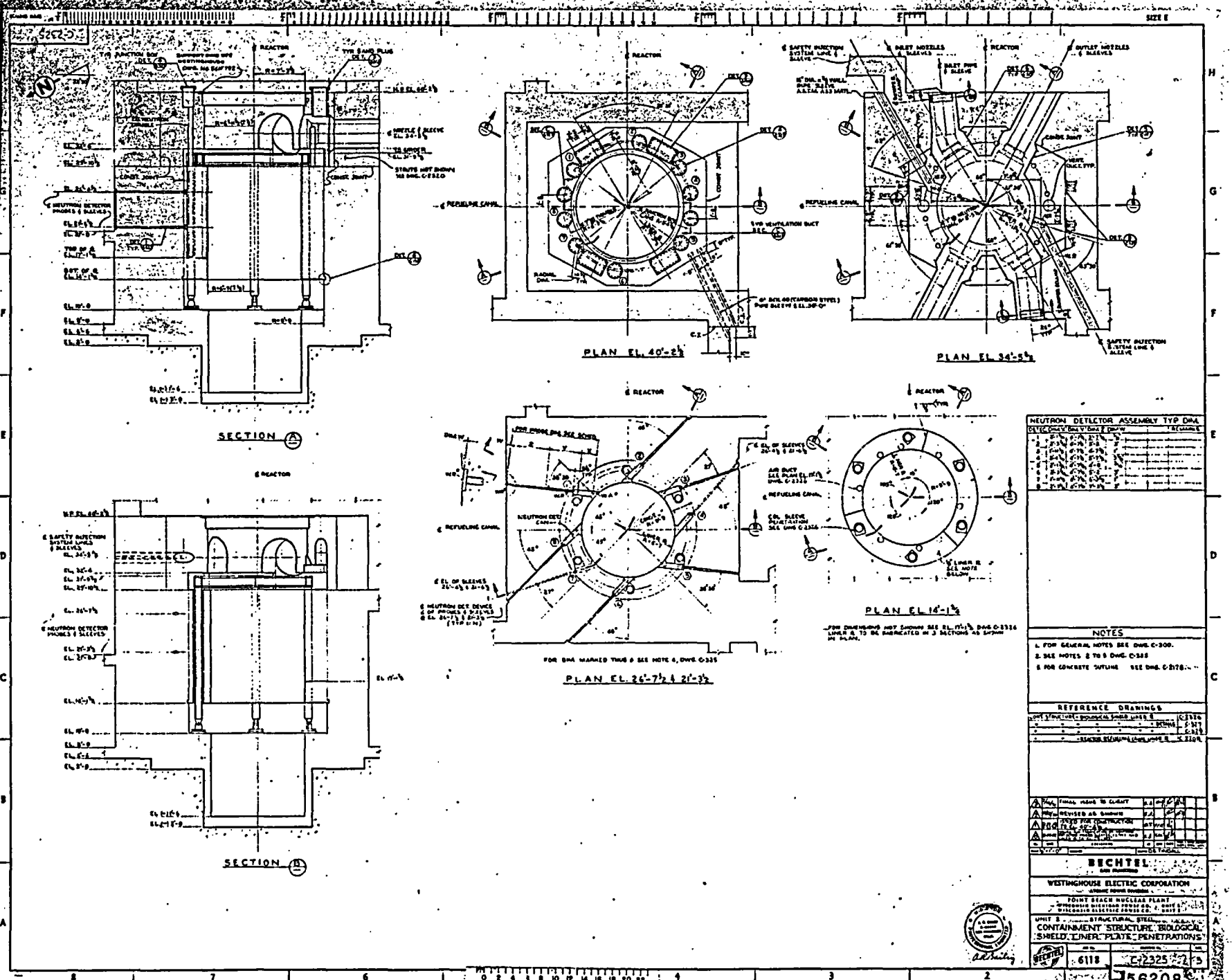


30XNOV1975

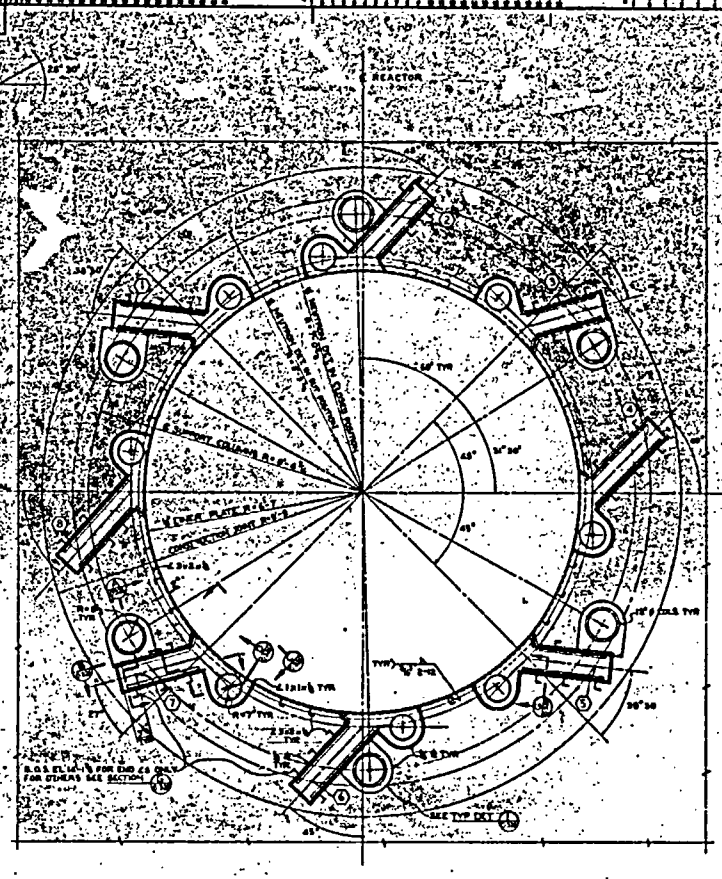
56162


PLAN EL 31'-9½"
PLAN @ EL 10'-0"
SECTION
DETAIL
ELEVATION
WESTINGHOUSE ELECTRIC CORPORATION
POINT BEACH NUCLEAR PLANT
REACTOR STEEL SUPPORTS
6118
C-2320
BECH 6118 C-2320
STEEL
REACTOR STEEL SUPPORTS
POINT BEACH N.P. UNIT 2
56202
05
E

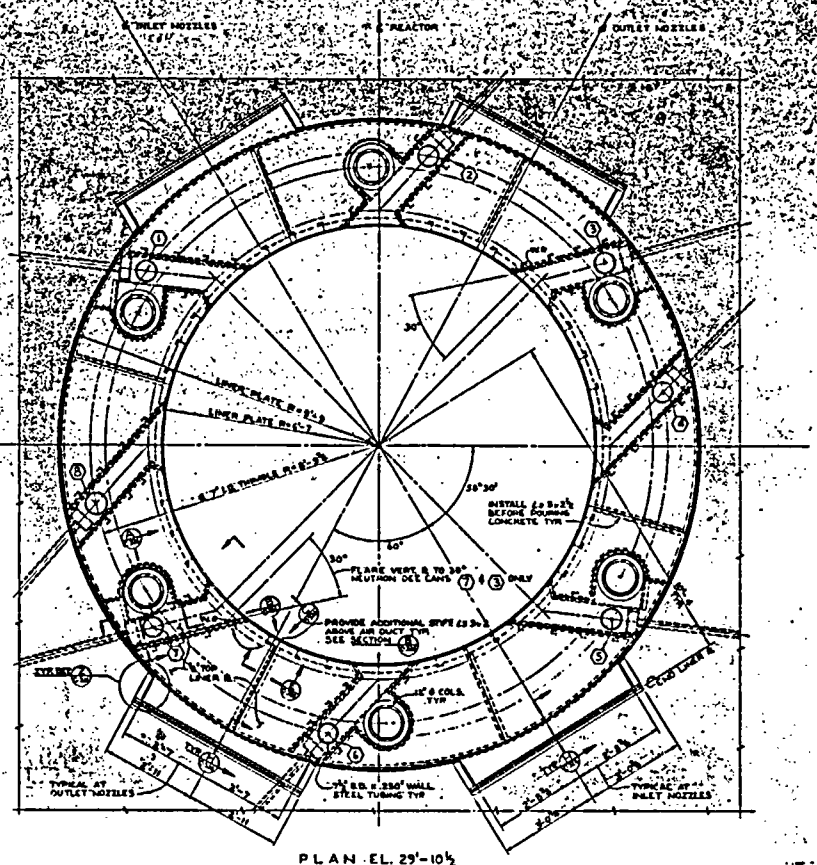

PLAN ELEVATION 10'-0"
LOCATION PLAN-LOWER SUPPORTS
LOCATION PLAN-INTERMEDIATE SUPPORTS
℄ REACTOR GIRDER
REFUELING CANAL
REACTOR 2-R1
REACTOR / STEAM GENERATOR
COOLANT PUMP CASING 2-P1B
PRIMARY LOOP SUPPORT PIER (TYP.)
REACTOR INLET NOZZLE
REACTOR OUTLET NOZZLE
REACTOR SUMP COLS
FOR DETAILS NOT SHOWN SEE DWGS C-2320 & C-2321
NOTES:
REFERENCE DWGS.
REACTOR SITEL SUPPORTS C-2320
STEAM GENERATOR (COOLANT PUMP SUPPORTS C-2321
STEAM GENERATOR (COOLANT PUMP SUPPORTS C-2322
STEAM GENERATOR SUPPORTS C-2323
STEAM GENERATOR SUPPORTS C-2324
BECHTEL
WESTINGHOUSE ELECTRIC CORPORATION
POINT BEACH NUCLEAR PLANT
UNIT 2
STEEL
LOCATION PLANS
MAJOR COMPONENT SUPPORT STRUCTURES
6118
C-2322
3
30XNOV1975
5'6204


SECTION A
SECTION B
PLAN EL. 40'-2½
PLAN EL. 34'-5½
PLAN EL. 26'-7½ & 21'-3½
PLAN EL. 14'-1½
NEUTRON DETECTOR ASSEMBLY TYP DWG
NOTES
1. FOR GENERAL NOTES SEE DWG. C-300.
2. SEE NOTES 2 TO 5 DWG. C-325
3. FOR CONCRETE OUTLINE SEE DWG. C-2178
REFERENCE DRAWINGS
BECHTEL
WESTINGHOUSE ELECTRIC CORPORATION
POINT BEACH NUCLEAR PLANT
CONTAINMENT STRUCTURE, BIOLOGICAL SHIELD, LINER PLATE, PENETRATIONS
6118
C-2325
56208

30XNOV1975



PLAN EL. 17'-1⅝
¼"=1'-0"



PLAN EL. 29'-10½
¼"=1'-0"

FOR DWG NOT SHOWN SEE PLAN EL. 17'-1⅝

NOTES

1. FOR NOTES SEE DWG. C-325 & C-326

1	10-9	FINAL ISSUE TO CLIENT			
1	10-9	ISSUED FOR CONSTRUCTION			
1		ISSUED FOR FABRICATION			

BECHTEL

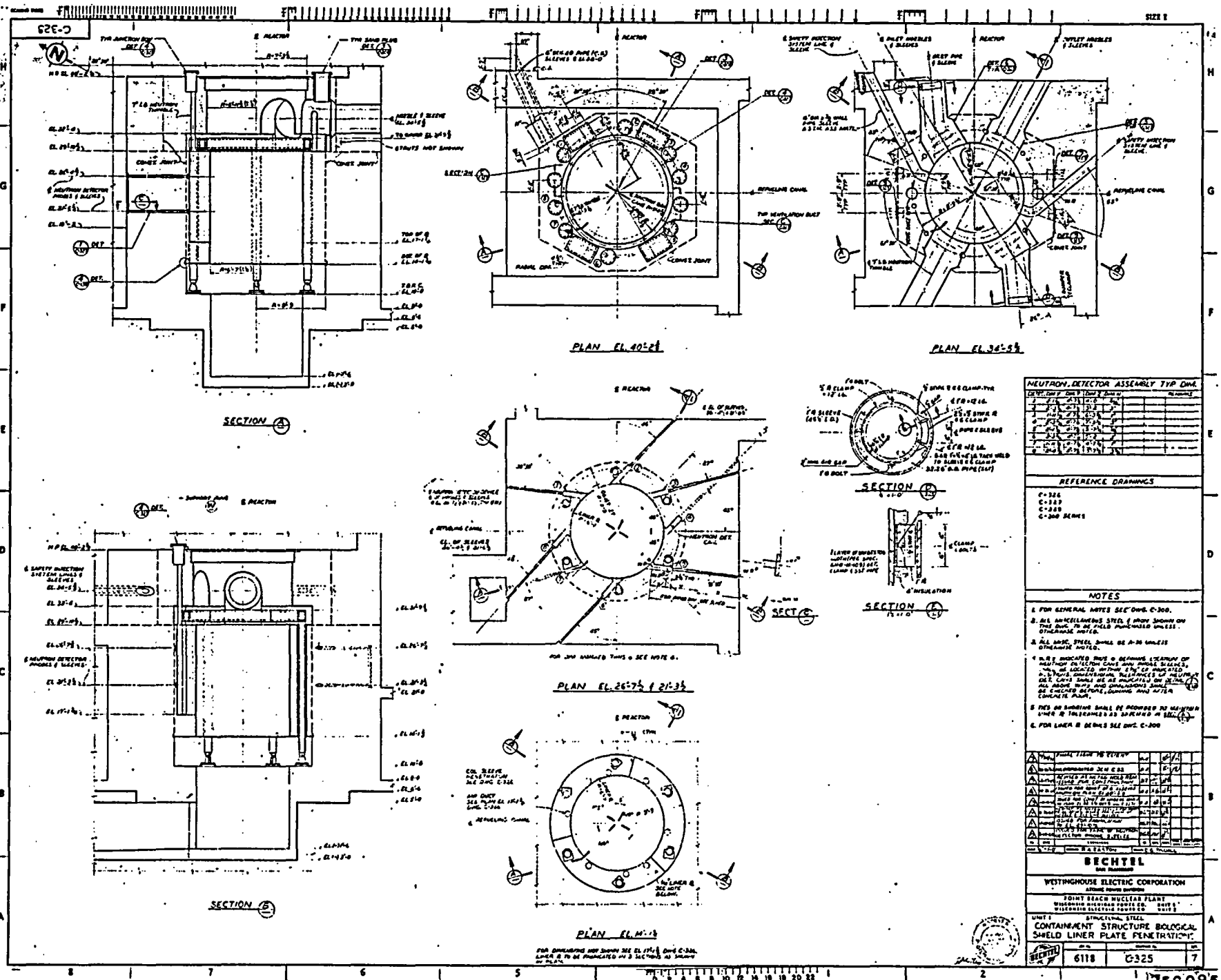
WESTINGHOUSE ELECTRIC CORPORATION

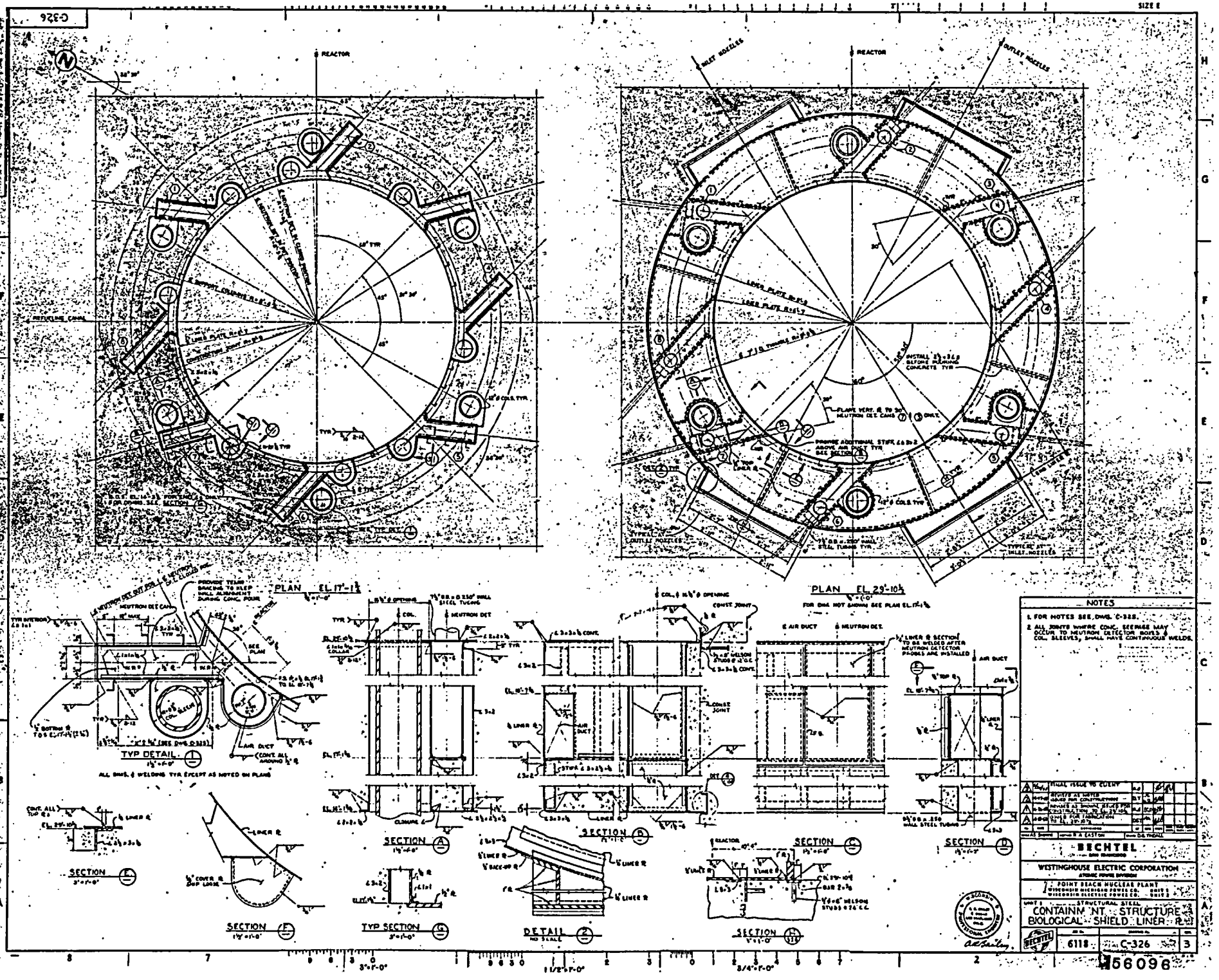
POINT BEACH NUCLEAR PLANT
WISCONSIN MICHIGAN POWER COMPANY
WISCONSIN ELECTRIC POWER COMPANY

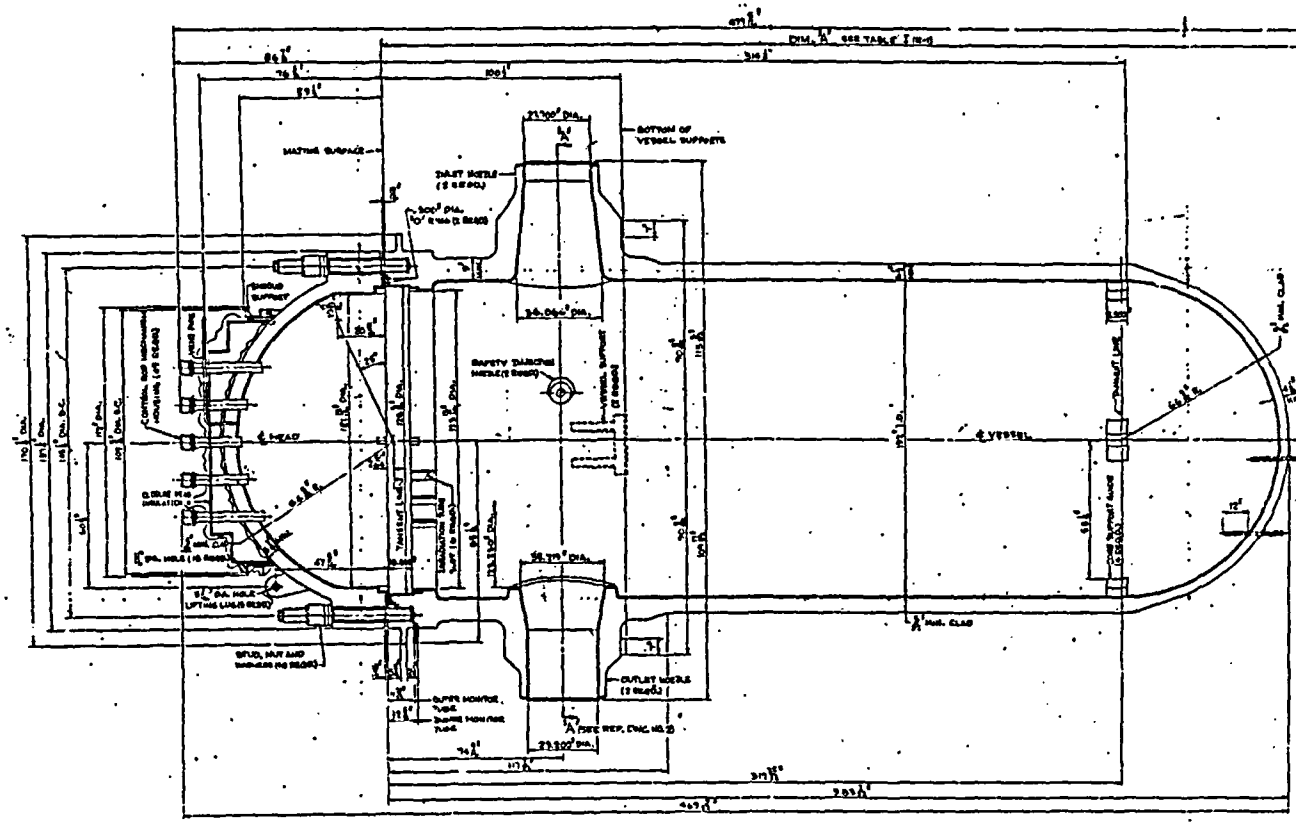
UNIT 2 — STRUCTURAL STEEL

CONTAINMENT STRUCTURE
BIOLOGICAL SHIELD & LINER PLATE

6118 — C-232


C-325
SECTION
PLAN EL. 40'-2½
PLAN EL. 34'-5½
SECTION
SECTION
SECTION
PLAN EL. 26'-7½ & 21'-3½
PLAN EL. 14'-1½
NEUTRON DETECTOR ASSEMBLY TYP. DWG.
REFERENCE DRAWINGS
C-326
C-327
C-328
C-300 SERIES
NOTES
1. FOR GENERAL NOTES SEE DWG. C-300.
2. ALL MISCELLANEOUS STEEL & PLATE SHOWN ON THIS DWG. TO BE FIELD PURCHASED UNLESS OTHERWISE NOTED.
3. ALL MISC. STEEL SHALL BE A-36 UNLESS OTHERWISE NOTED.
6. FOR LINER & DETAILS SEE DWG. C-300
BECHTEL
WESTINGHOUSE ELECTRIC CORPORATION
POINT BEACH NUCLEAR PLANT
WISCONSIN MICHIGAN POWER CO. UNIT 1
WISCONSIN ELECTRIC POWER CO. UNIT 2
STRUCTURAL STEEL
CONTAINMENT STRUCTURE BIOLOGICAL SHIELD LINER PLATE PENETRATION
6118
C-325
7
30XNOV1975
356095


PLAN EL 17'-1½
PLAN EL 29'-10½
TYP DETAIL
SECTION
TYP SECTION
DETAIL
NOTES
1. FOR NOTES SEE DWG. C-325.
2. ALL JOINTS WHERE CONC. SEEPAGE MAY OCCUR TO NEUTRON DETECTOR BOXES & COL. SLEEVES, SHALL HAVE CONTINUOUS WELDS.
BECHTEL
WESTINGHOUSE ELECTRIC CORPORATION
POINT BEACH NUCLEAR PLANT
STRUCTURAL STEEL
CONTAINMENT STRUCTURE
BIOLOGICAL SHIELD LINER
6118
C-326
3
456096



SECTIONAL ELEVATION

TABLE I (N-6)	
PENETRATION NUMBER	DIMENSION "L"
68	590
68A	590
68B	590
67A AND 67B	580
66A AND 66B	566
65	567
70A AND 70B	564
70C AND 70D	564
71	565
72	564
75A AND 75B	592
74A AND 74B	568
78	581
76	581
77	581
79A AND 79B	577
79	577
80	576
81	576
82	577
83	576
84	576
85	576
86	576
87	576
88A AND 88B	576
89A AND 89B	576

* SEE TRAVELER DWGS 233-681S-01, -02, -03, AND -04 FOR CLOSURE HEAD INSULATION. (SEE 01-121)

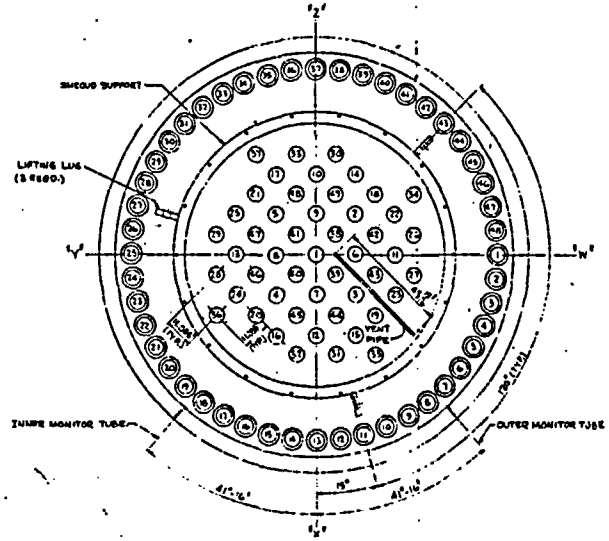
THIS DRAWING HAS BEEN SCANNED
FROM CE
DWG. NO. 233-681 REV. 00
DATE 11-13-00 BY DEAF ACT. No. 3063

GENERAL NOTES
1. FOR STANDARD NOTES SEE REF. DWG. NO.1.
2. ALL DIMENSIONS ARE REFERENCE.
3. ALL FABRICATION SHALL BE IN ACCORDANCE WITH ASME CODE SECTION III.
4. DESIGN PRESSURE: 2,485 P.S.I.G.
5. DESIGN TEMPERATURE: 650°F.

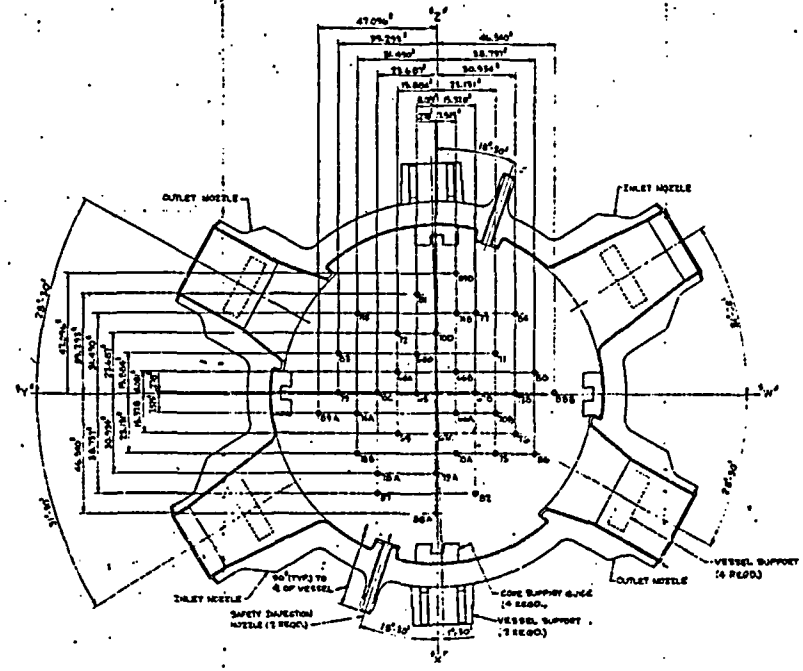
REFERENCE DRAWINGS

NO.	TITLE	DWG. NO.
1	STANDARD NOTES	A-341-101
2	GENERAL ARRANGEMENT PLAN	A-341-102
3	REACTOR VESSEL TOP MACHINING	A-341-103
4	CLOSURE HEAD TOP MACHINING	A-341-104
5	CLOSURE HEAD STUDS	A-341-105

GENERAL ARRANGEMENT "ELEVATION"
FOR: WESTINGHOUSE ELECTRIC CORP.
1411 S.D. P.W.R.
E 233-681 0
4869
CE 233-681
17903
1017306
GENERAL ARRANGEMENT "ELEVATION"
POINT BEACH N.P. UNIT 2
EPB 02 MRCL199004001
13-NOV-2000



PLAN VIEW



SECTION A-A (SEE REF. DWG. NO. 2)

30X JUNE 1974

GENERAL NOTES

1. FOR STANDARD NOTES SEE REF. DWG. NO. 1.
2. ALL DIMENSIONS ARE REFERENCE.
3. ALL FABRICATION SHALL BE IN ACCORDANCE WITH ASME CODE SECTION III.
4. DESIGN PRESSURE 2485 P.S.I.G.
5. DESIGN TEMPERATURE 650° F.

REFERENCE DRAWINGS

NO.	TITLE	DWG. NO.
1.	STANDARD NOTES	A-245-381
2.	GENERAL ARRANGEMENT "ELEVATION"	E-233-681
3.	PRESSURE VESSEL FINAL MACHINING	E-233-685
4.	CLOSURE HEAD FINAL MACHINING	E-233-683
5.	CLOSURE HEAD ASSEMBLY	E-233-690

COMBUSTION ENGINEERING, INC., WINDSOR, CONN.

3/4" = 1'-0"

CONTRACT 4869

DWN. BY J.R. Simmons, Jr.

COMBUSTION ENGINEERING, INC.
CHATTANOOGA DIVISION

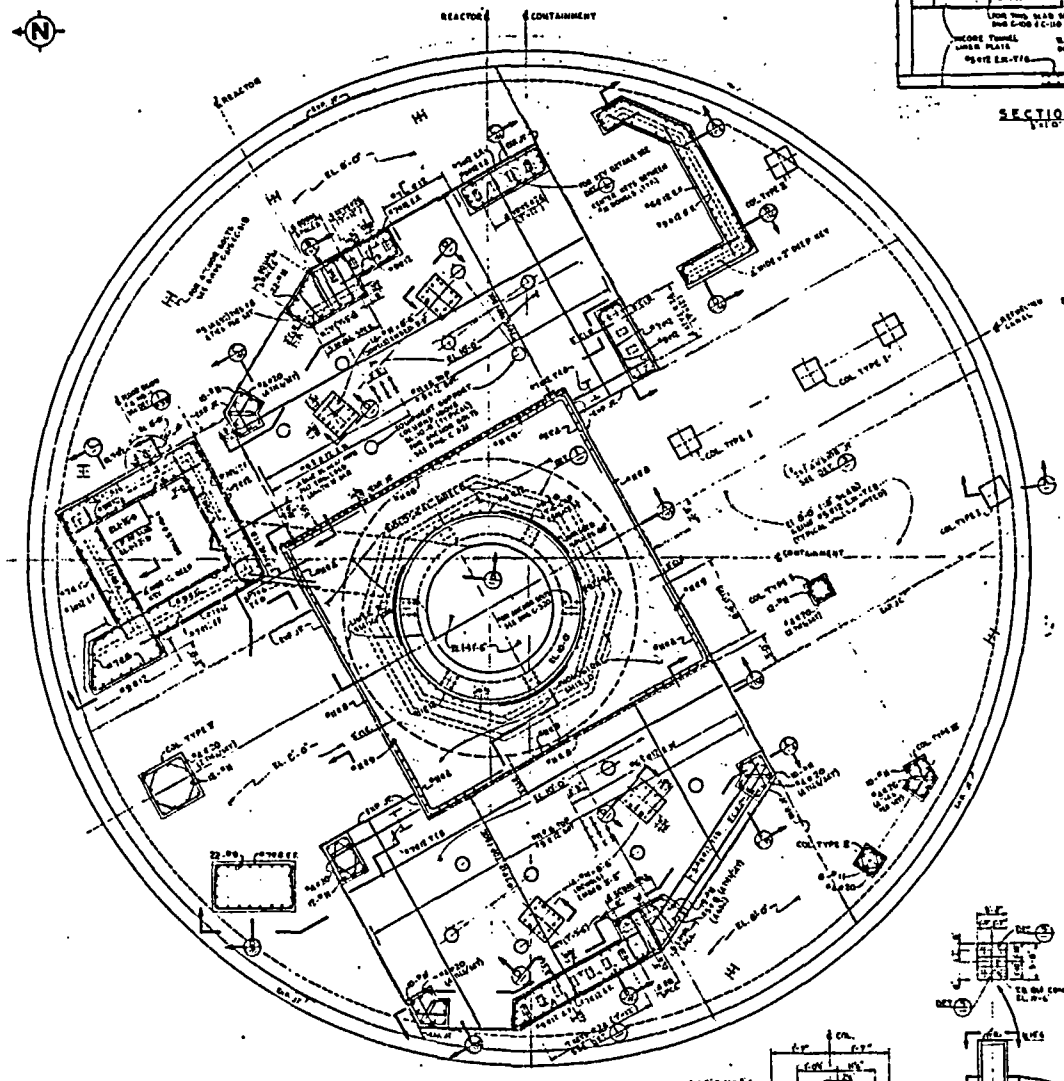
GENERAL ARRANGEMENT "PLAN"

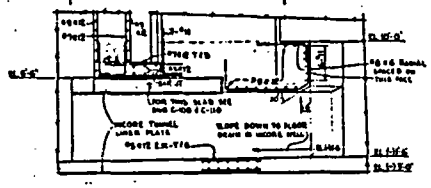
FOR: WESTINGHOUSE ELECTRIC CORP.
173" I.D. P.W.R.

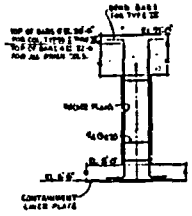
E 233-682 0

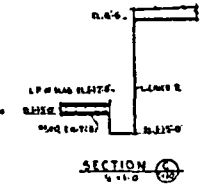
17904

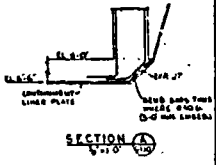
C-130

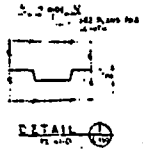


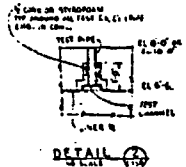


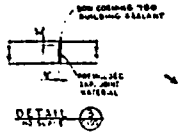


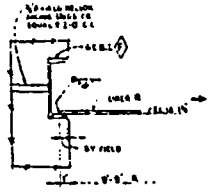


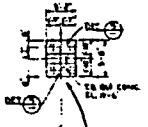


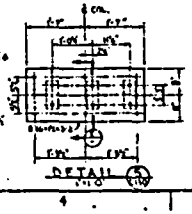


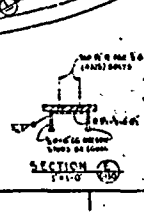












NOTES

1. FOR CONCRETE OUTLINE, NOTES AND REFERENCE DRAWINGS SEE DWG. C-125
2. FOR TYPICAL DETAILS SEE DWG. C-105
3. THE EXPANSION JOINT SEALANT WAS REPLACED WITH DOW CORNING 888 SEALANT DURING JOB PER MCR 88-026. THESE WERE SEAL 0" CORE HOLES DRILLED IN THE CONCRETE SLAB TO INSPECT LINER PLATE. REVIEW MCR 88-026 FOR ADDITIONAL INFORMATION.

BECHTEL
SAN FRANCISCO

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIVISION

WISCONSIN MICHIGAN POWER COMPANY
POINT BEACH NUCLEAR PLANT UNIT NO. 2

CONCRETE
CONTAINMENT STRUCTURE INTERIOR
REINFORCING PLAN EL 10'-0" & DETAILS

6118	C-130	6

55969

P8015020002007 0 CRBailey

ENCLOSURE 5

WESTINGHOUSE AUTHORIZATION LETTER
AFFIDAVIT
PROPRIETARY INFORMATION NOTICE
COPYRIGHT NOTICE

(21 pages follow)

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4643
Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-05-2004

June 9, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WEP-05-187, "Point Beach Unit 2 Reactor Vessel CMTRs" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced letter and its attachment is further identified in Affidavit CAW-05-2004 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2004, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney, NRC
L. Feizollahi, NRC

CAW-05-2004

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

PATRICIA L. CROWN
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 9th day
of June, 2005

Patricia L Crown
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Patricia L. Crown, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Feb. 7, 2009
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the attachment to WEP-05-187, "Point Beach Unit 2 Reactor Vessel CMTRs" (Proprietary), dated June 9, 2005. The information is provided in support of a submittal to the Commission, being transmitted by Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Point Beach Unit 2 contains design information that is proprietary to Westinghouse and is provided in response to certain NRC requirements for justification of reactor vessel head drop analyses.

This information is part of that which will enable Westinghouse to:

(a) Show that a postulated drop of the replacement reactor vessel closure head would produce impact forces at the vessel supports that are no greater than those calculated for the original vessel head, accounting for the different weights of new replacement reactor vessel head and head assembly upgrade packages.

(b) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of this information to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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