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NLS2007012
April 17, 2007

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to Request for Additional Information Regarding License Amendment Request for Onsite Spent Fuel Storage Expansion
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:
1. Letter from Carl F. Lyon, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, dated March 5, 2007, "Cooper Nuclear Station-Request for Additional Information Re: License Amendment Request-Onsite Spent Fuel Pool Storage Expansion (TAC NO. MD3349)"
2. Letter from Stewart B. Minahan, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated October 17, 2006, "License Amendment Request to Revise Technical Specification-Onsite Spent Fuel Storage Expansion" (NLS2006028)

The purpose of this letter is for the Nebraska Public Power District (NPPD) to submit a response to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by NRC letter dated March 5, 2007 (Reference 1). The additional information in this letter is to support the NRC review of the license amendment request for onsite spent fuel storage expansion at the Cooper Nuclear Station submitted by NPPD letter dated October 17, 2006 (Reference 2).

The Reference 1 letter provided questions from the Health Physics Branch (IHPB), the Engineering Mechanics Branch (EEMA), and the Steam Generator Tube Integrity and Chemical Engineering Branch (CSGB). The response to the questions from the IHPB is provided in Attachment 1. The response to the questions from the EEMA is provided in Attachment 2. The response to the questions from the CSGB is provided in Attachments 3 and 4.

The information submitted by this RAI response does not change the evaluation of the No Significant Hazards Consideration submitted by the Reference 2 letter.

Should you have any questions regarding this submittal, please contact Paul Fleming, Licensing Manager, at (402) 825-2774.

A001

I declare under penalty of perjury that the foregoing is true and correct.

Executed on APR 17 2007

Sincerely,

Stewart B. Minahan
Vice President-Nuclear and
Chief Nuclear Officer

/rr

Attachments

Enclosure

cc: Regional Administrator w/ Attachments
USNRC - Region IV

Cooper Project Manager w/ Attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/ Attachments
USNRC - CNS

Nebraska Health and Human Services w/ Attachments
Department of Regulation and Licensure

NPG Distribution w/o Attachments

CNS Records w/ Attachments

Attachment 1

Response to Health Physics Branch Request for Additional Information Regarding License Amendment Request for Spent Fuel Pool Storage Expansion

Cooper Nuclear Station, Docket No. 50-298, DPR-46

1. NRC Request

In Section 9.4 of the Holtec Report, you state that the calculated dose rate adjacent to the new fuel racks (if they are completely filled with freshly discharged fuel) would be unacceptably high.

(a) *Provide the current radiation dose rates for those accessible areas adjacent to the area (including any accessible areas below the spent fuel pool [SFP]) where the new fuel racks will be installed.*

(b) *For these same areas, provide the calculated dose rates from the proposed new racks (1) if the new racks are completely filled with freshly discharged fuel and (2) if the outer rows of the new racks are loaded with fuel which has decayed for at least 5 years.*

(c) *Provide the number of outer rows which will be filled with aged fuel to ensure that the dose rates in potentially-occupied areas adjacent to the SFP do not exceed the current radiation dose rates for these areas.*

NPPD Response

(a) The general area dose rates in areas adjacent to the spent fuel pool (SFP) at Cooper Nuclear Station (CNS) are typically less than 2 millirem/hour (mr/hr). (This is the dose rate at three feet from the walls. The dose rate on contact with the surface of the wall is not taken). Accessible areas adjacent to the SFP, including areas below the SFP, are surveyed quarterly.

(b) The following are the calculated dose rates with spent fuel in the new racks.

(1) With the new racks completely filled with freshly discharged fuel, the maximum dose rate on contact with the surface of the SFP wall would be 278 mr/hr.

(2) With the new racks completely filled with freshly discharged fuel, with 5-year old decayed fuel in the two outer rows along the sides of the new racks facing the

pool walls, the maximum dose rate on contact with the surface of the SFP wall would be 1.7 mr/hr.

(c) Two rows of 5-year cooled fuel will be placed along the sides of the new racks facing the fuel pool walls. (This information modifies the commitment made in the initial license amendment request, submitted by letter dated October 17, 2006, to read as follows: "Two rows of 5-year cooled fuel will be placed along the sides of the new racks facing the fuel pool walls to provide shielding from freshly discharged fuel assemblies. The procedure for controlling storage of spent fuel in the spent fuel pool will be revised to require the placement of two rows of 5-year cooled fuel.")

2. NRC Request

Provide the expected maximum dose rate above the SFP.

NPPD Response

The area above the SFP was surveyed to address this question. (The area above the SFP is not routinely surveyed for dose rates since this area is not typically occupied, but is monitored during refueling operations when personnel are working above the SFP). This survey showed that the dose rates at the surface of the water ranged from 2 mr/hr to approximately 8 mr/hr.

Chapter 9, "Radiological Evaluation," of the Holtec Licensing Report (proprietary and non-proprietary versions were provided as Enclosures 1 and 4, respectively, of the initial amendment request dated October 17, 2006), states that there is essentially no change in the maximum dose rate above the SFP due to the increased storage capacity. The basis for this conclusion, stated in the report, was that the difference in cell pitch between the existing racks and the new racks is minimal.

Therefore, the maximum expected dose rate above the SFP following placement of the additional fuel assemblies is the current highest measured dose rate of approximately 8 mr/hr.

3. NRC Request

Describe any radiation surveys that will be performed (from the pool rim or by divers in the pool) to map dose rates in the SFP prior to dive operations.

NPPD Response

Dive operations in the SFP to prepare the SFP for placement of the additional racks were completed in August 2006. At this time no additional dive operations in the SFP are planned.

The following is a discussion of the radiation surveys that would be performed in the event that additional dive operations are needed.

Prior to divers entering the SFP, underwater surveys of the areas involved in the dives are performed to map out the areas. These surveys consist of contact and general area readings in the diver's work area and travel path to and from the work area. At a minimum, surveys include dose rate readings, including details of radiation field gradients, at the floor and walls of the SFP, the top surface of the spent fuel racks, underwater fixtures, work tables, and pipes.

The need to perform additional surveys is evaluated (1) prior to the first dive of the day if diving operations had been suspended, (2) when changing conditions are expected, (3) if dose readings are irregular.

These surveys are performed using Merlin Gerin AMP-100 or AMP-200 underwater radiation survey instruments. These are hand-held microprocessor based Geiger-Mueller type gamma survey instruments. The AMP-100 instrument measures dose in the range of 0.050 Rem per hour (R/hr) to 999.9 R/hr. The AMP-200 instrument measures dose in the range of 5.0 R/hr to 15,000 R/hr. Both instruments allow selection of different ranges (threshold values) of dose.

These underwater surveys are performed from the rim of the SFP and/or the refueling bridge with the underwater instrument affixed to the end of a pole. Additional surveys are performed as the diving operations progress.

The diver is also equipped with an underwater probe having remote readout capability. An individual from the Radiological Protection (RP) department monitors the readout on this instrument from outside the SFP. The divers use this instrument to survey their work area while in the SFP. Finally, teledosimetry is used to monitor the diver, thereby providing a continuous measurement and readout to the RP individual outside the SFP of the dose rate and accumulated dose to the diver.

4. NRC Request

(a) *Provide a description of any sources of high radiation that may be in the SFP during diving operations for minor modification of the beam segments.*

(b) *Discuss what precautions (such as use of TV monitoring, additional lighting, tethers, use of physical barriers between the divers and high radiation sources, fuel shuffling, etc.) will be used to ensure that the divers will maintain a safe distance from any high radiation sources in the SFP.*

(c) *Discuss how you plan to alert divers (e.g., by use of continuous voice communication with the divers) in the event that they inadvertently enter an area of high radiation or if they exceed their predetermined dose limit while performing work in the SFP.*

(d) *Discuss the use of multiple dosimeters on divers who will be working in the SFP.*

(e) *Discuss how divers will be monitored for contamination when they exit the SFP.*

NPPD Response

As noted in the response to question 3 above, dive operations in the SFP were completed in August 2006, and no additional dive operations in the SFP are planned at this time. The following responses reflect what would be done during any future dive operations if determined to be needed.

(a) The items stored in the SFP that would be considered as sources of high radiation, and that might result in a significant dose to a diver in the pool, are the spent fuel assemblies stored in the existing racks, used control rod blades, and several filters from previous vacuuming operations stored in the northwest corner of the SFP.

(b) The following procedural precautions and provisions are provided to ensure that divers will maintain a safe distance from high radiation sources.

    (1) A minimum separation of 10 feet is to be maintained between the diver and any spent fuel assembly, control component, or irradiated component.

    (2) Spent fuel or other highly radioactive components shall not be moved in the diving area while dive operations are in progress. (A "Highly Radioactive Component" is defined as a component within the dive area having a contact dose rate greater than 1,000 mr/hr).

    (3) A "Safe Dive Zone" must be established for areas in the SFP containing any spent fuel assembly, control component, irradiated component, or highly radioactive components. A "Safe Dive Zone" is an area established to ensure that the diver is protected from coming in contact with the fuel assemblies or components.

(4) Highly visible, physical boundaries are required to be used in areas of the SFP containing Highly Radioactive Components.

(5) A pool-side briefing is required prior to starting the dive operation to ensure that all safety precautions are taken and to discuss radiation levels in the work area with the diver, and to indicate areas of the SFP with a low dose rate that are considered suitable for waiting when not actively engaged in work.

(c) The following procedural provisions are specified to monitor the dose rate in areas in which the divers are working and to alert the divers if their dose limits are exceeded.

(1) Perform a communications check prior to starting dive operations to ensure that both the diver and the control station can both receive and transmit messages.

(2) Continuously monitor radiation levels in the dive zone and surface dose rates during diving operations.

(3) Communicate the dose rate to the diver during diving operations.

(4) Continuously monitor telemetry on the diver during diving operations.

(d) The following are the procedural provisions for the use of multiple dosimeters on divers.

(1) Multiple dosimetry should be utilized for diving operations involving areas of the SFP that contain Highly Radioactive Components. The procedure specifies that dosimetry shall be affixed to the diver's chest, both arms, and both thighs.

(2) A survey instrument with remote readout capability is to be attached to the diver.

(e) The following are the procedural provisions for monitoring divers for contamination when exiting from the SFP.

(1) The divers are to be washed down with water as they exit the SFP. (The water to be used must be approved for this purpose).

(2) After the washdown the dive suit is surveyed for hot spots or hot particles using an open window ion chamber prior to removing the suit from the diver. If a hot spot or a hot particle is detected the wash down of the diver is to be repeated immediately.

(3) Following the dives the dive equipment and the suit worn by the diver are surveyed for contamination. CNS RP performs or observes a survey of the entire body of the diver.

5. NRC Request

In section 9.4 of the Holtec Report, you state there are continuous airborne monitors for airborne activities available in the immediate vicinities of the pool. Describe how these monitors are utilized in the SFP area during the Refueling Floor Operations, including diving operations and SFP rack installations.

NPPD Response

There is one Continuous Air Monitor (CAM) on the north end of the refuel floor that is normally in service. This CAM monitors airborne radioactivity from both iodines and particulates. Another CAM is located on the south end of the refuel floor, adjacent to the SFP. Normal practice is to place this CAM in service during diving operations and refueling outages (i.e., when spent fuel is being placed in the SFP). Personnel in the areas being monitored by the CAM are instructed to leave the area in the event that the CAMs alarm. In addition to these CAMs, low volume air samples are taken during any work on the refuel floor.

6. NRC Request

Discuss the removal, decontamination, and disposal of the storage racks for the control rod blades and the drum platform from the shipping cask storage area.

NPPD Response

The drum platform has been removed from its location in the SFP and is temporarily located in the cask pit area in the SFP. This platform will be removed from the SFP and shipped offsite for burial. This will be done at the time that the new rack A is installed in the SFP, if not done sooner.

The control rod blade storage rack was relocated to the cask pit area from the area north of the cask pit area in the SFP. This rack is needed for efficiency of refueling outages and will remain in the spent fuel pool. It will be temporarily removed from the SFP if Rack B is installed or during Dry Fuel Storage loading campaigns.

The specific activities that will be used to decontaminate the drum platform and the control rod blade storage rack when removed from the SFP have not been definitively established at this time. However, good practices to ensure that doses are maintained As Low as Reasonably Achievable will be used during the removal and decontamination of the platform. These good practices include, but are not limited to, the following:

a) Continuously monitoring the dose rate near the SFP while the platform is being raised in the SFP.

b) Rinsing the platform with demineralized water and/or hydrolazing the platform as it breaks the surface of the pool during removal to minimize airborne contamination.

c) Wrapping the platform in plastic to minimize airborne contamination.

In addition, the platform will be prepared for offsite shipping in accordance with applicable regulations.

7. <u>NRC Request</u>

Discuss how the storage of the additional spent fuel assemblies in the SFP will affect the releases of radioactive gases from the SFP. Specifically address the potential increased releases of Kr-85, I-131, and tritium from the SFP.

<u>NPPD Response</u>

Currently no radioactive gases are released from the SFP. The increased number of fuel assemblies in storage as a result of the increased SFP capacity is not expected to create the potential for release of radioactive gases from the SFP. This conclusion includes consideration of the radioactive gases Kr-85, I-131, and tritium.

8. <u>NRC Request</u>

Discuss how the storage of additional spent fuel assemblies in the SFP will affect the release of radioactive liquids from the plant as a result of any expected increase in the change out frequency of the demineralizer resins for the SFP cleanup.

<u>NPPD Response</u>

The increased number of fuel assemblies stored in the SFP is not expected to result in an increase in the radionuclide concentrations in the SFP water. As a result, no increase in the frequency of changing the demineralizer resins or in the amount of spent resins is expected.

Attachment 2

Response to Engineering Mechanics Branch Request for Additional Information Regarding License Amendment Request for Spent Fuel Pool Storage Expansion

Cooper Nuclear Station, Docket No. 50-298, DPR-46

1. NRC Request

As noted in Section 1.3 of the Holtec report, "The existing swing bolts are used to fasten the platform to the pool floor rendering them into a fixed appurtenance to the pool slab".

Please provide a summary of the stress analysis of the swing bolts and swing bolt anchorages to the pool slab for the loads transmitted by the Rack A fuel rack and platform. Please provide a drawing showing the anchorage pattern.

NPPD Response

The swing bolts and swing bolt anchorages have been analyzed for the loads transmitted by the Rack A fuel rack and platform to demonstrate that the stress levels meet the applicable ASME Section III, Subsection NF limits. The stress levels have been determined based on strength of materials formula. The following table summarizes the results for the swing bolts and swing bolt anchorages.

Item	Calculated Stress	Allowable Stress	Safety Factor
Anchorage Weld Stress	1600 psi	4000 psi	2.5

Burns and Roe drawings Number 4228, Rev. 11, and Number 4230, Rev. 14, are provided as part of this response. The locations of the hold-down bolts are shown on Drawing No. 4228, in the section labeled *Plan – Fuel Storage Pool @ El. 962'3"*. The dashed lines running East-to-West (North is shown on the drawing) show the location of the embedded "W" sections to which the ¼-inch stainless steel (SS) liner plate pieces were welded. The small dots on these lines depict the location of the hold-down bolts ("swing-bolts") which were welded directly to the same embedded "W" sections, and to the ¼-inch SS liner plates. The anchorage pattern is shown in Figure 1.

Drawing No. 4230, Sections 859-859 and 860-860, shows the details of the welded connection between the swing-bolt clevis and the embedded "W" section/liner plate. The connection of the "W" sections to the cast-in-place anchors is shown in Section 846-846.

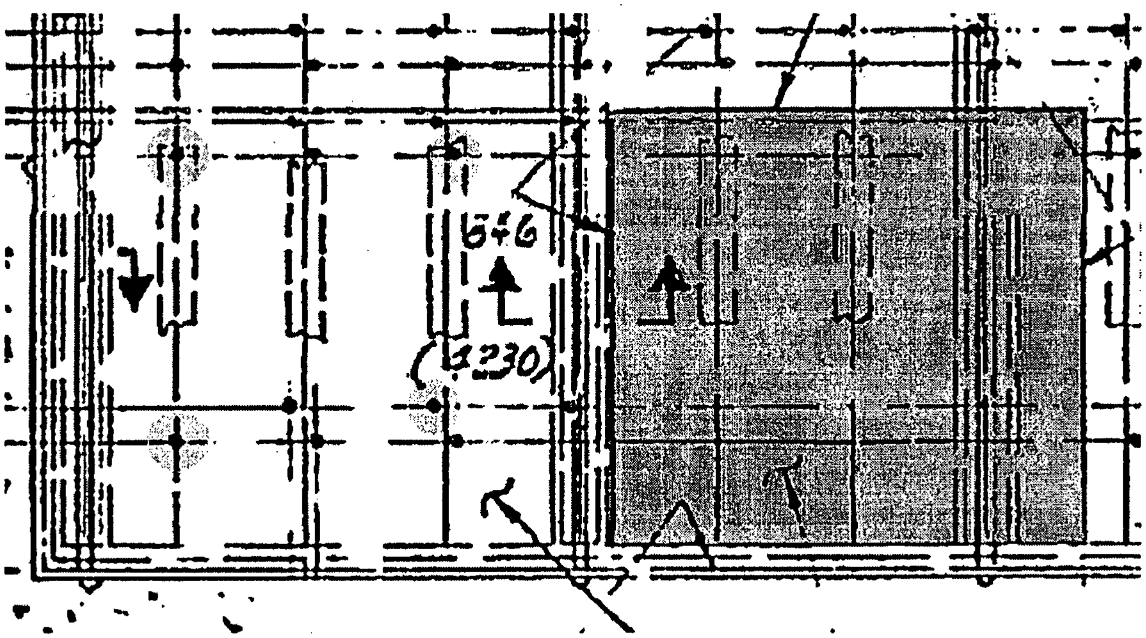


Figure 1

NOTE
This figure is an excerpt from the structural detail information (section labeled *Plan - Fuel Storage Pool @ EL. 962'-3"*) provided on Burns and Roe Drawing 4228. The anchorage pattern of the four (4) "swing-bolts" utilized to secure the platform for new Rack A is shown by the light grey shading. For reference purposes, the shaded area is the existing pool-floor cask bearing pad (1-inch plate).

2. NRC Request

Section 2.1 of the Holtec report notes that: "Although both rack modules are freestanding, and thus may slide during a seismic event, any significant membrane strains in the pool liner are prevented by the presence of the platforms. As a result, the maximum strain sustained by the liner during a seismic event is assumed to be less than the ultimate strain for the liner material (austenitic stainless steel, ultimate strain ≥ 0.38)."

Please provide a summary of the stress analysis of the 2 inch thick by 2-1/2 inch wide "cask restraint ring" shown in Section 3711 of Burns and Roe Drawing 4288 for the loads induced by Rack B.

NPPD Response

Section 3711-3711 of Burns and Roe Drawing Number 4288, Rev. 1, (provided in support of this response) shows the detail of the configuration of how the "cask restraint ring" is welded to the 1-inch thick cask pad, which is welded to the 1/4-inch SS liner plate. The 2-inch thick by 2-1/2 inch-wide "cask restraint ring", shown in Section 3711-3711, is secured to the 1-inch thick cask bearing pad by a continuous 5/16-inch fillet weld. The fillet weld is located on the outside of the ring, which has an outer diameter of 83 inches (6'-11"). Thus, the total "length" of the subject weld is approximately 260 inches.

The platform for Rack B does not bear vertically upon the "cask restraint ring", nor does the platform for Rack B contact the 1/4-inch pool liner. All vertical bearing loads are transferred directly to the 1-inch thick cask bearing pad. The "cask restraint ring" can only be postulated to carry lateral, shear-type loadings caused by the Operating Basis Earthquake (OBE) and/or the Safe Shutdown Earthquake (SSE).

The "cask restraint ring" and bearing pad plate material is American Society for Testing and Materials (ASTM) A-240, Type 304 stainless steel, which has a specified minimum yield strength, S_y, of 21,300 psi, and a specified minimum tensile strength, S_u, of 66,200 psi (both values are at 200°F).

The Service Level A allowable shear stress limit in the base material is 0.40 S_y. The Service Level D allowable shear stress limit in the base material is the *lesser* of 0.72 S_y and 0.42 S_u (which in this case is the 0.72 S_y value).

The weld material is conservatively assumed to have a tensile strength (S_u) of 60,000 psi. This assumes the use of E60XX electrodes for the installation of the "*cask restraint ring*", which is a low-strength grade of electrode.

The Service Level A allowable shear stress limit in the weld throat is 0.30 S_u. The Service Level D allowable shear stress limit in the weld throat is 0.42 S_u.

Therefore, the welded joint has the following approximate allowable shear load limits:

Service Level	Base Material	Weld Throat
A (applicable to OBE)	2,662.5 lb/inch	3,977.5 lb/inch
D (applicable to SSE)	4,792.5 lb/inch	5,568.5 lb/inch

Note that the shear load on the Base Material represents the critical stress condition.

When conservatively considering only approximately 25 percent of the available "length of weld" actually provided by the installed configuration is "active" in resisting applied lateral loadings, the cask restraint ring can be concluded to withstand the following "allowable" lateral shear-type loads:

Service Level	Base Material Allowable Shear Load
A (applicable to OBE)	173,000 pounds
D (applicable to SSE)	311,500 pounds

Note that the above allowable lateral/shear load values conservatively neglect any contribution from frictional resistance between the platform and the 1-inch thick cask bearing pad.

In comparison, the Platform for Rack B was found to have the following maximum potential applied lateral loadings transmitted from the storage rack:

Service Level	Rack B Lateral Load (Based on Table 6.8.1 of Holtec Report HI-2043224)
A (applicable to OBE)	98,700 pounds
D (applicable to SSE)	119,000 pounds

Therefore, the calculated lateral seismic load represents approximately 57 percent of the conservatively-determined "allowable" lateral load the configuration can withstand under OBE conditions, and, approximately 38 percent of the conservatively determined "allowable" lateral load the configuration can withstand under SSE conditions.

3. NRC Request

Section 2.2 of the Holtec report notes that: "Because the platforms are not an integral part of the rack, their stress analysis and structural qualification are not addressed in this licensing report".

Please provide a summary of the stress analyses for the Rack A and B platforms. Please provide the drawings for the platforms.

NPPD Response

A stress analysis of the Rack A platform has been performed using the finite element code ANSYS. The model, which is comprised of 4-noded shell elements, is shown in Figure 2. The finite element model is fixed at the base since the Rack A platform is restrained against lateral motion through its connection to four existing swing bolts on the SFP floor. Each of the swing bolts is preloaded to 10,000 lbs when the platform is installed to increase the frictional resistance at the SFP liner/platform interface. The maximum pedestal loads (in the horizontal and vertical directions) from the rack dynamic analysis are applied to the top surface of the Rack A platform model as nodal forces. Stresses in the platform components and welds are shown to meet ASME, Section III, Subsection NF and Appendix F for Level A and Level D service loadings, respectively. The minimum safety factors for the Rack A platform are summarized in the following table.

Platform and Components		Safety Factor	
		Level A (OBE)	Level D (SSE)
Platform	Membrane	1.24	1.07
	Membrane plus Bending	1.84	1.18
	Bearing	26.0	N/A
Welds	Bracket	10.5	4.05
	Mainframe Panel	2.79	2.75
	Top Pad	1.63	1.74
	Swing Bolt Hold Down Plate	1.64	N/A

The support platform for Rack A is shown on Holtec Drawing 4732 (5 sheets). The support platform for Rack B is shown on Black and Veatch Corporation Drawing No. S6002.

The platform for Rack B was evaluated for the effects of the maximum postulated lateral and vertical loads resulting from the critically-loaded storage rack condition. The Load Cases evaluated include the effects of Dead Load, combined with the dynamic effects of the OBE and the SSE. The following maximum pedestal loads (in the horizontal and vertical directions) from the rack dynamic analysis were applied to the top surface of the Rack B platform.

Service Level	Rack "B" Lateral Loads (See Table 6.8.1 of Holtec Report HI-2043224)	Rack "B" Vertical Loads (See Table 6.8.1 of Holtec Report HI-2043224)
A (applicable to OBE)	98,700 pounds	183,000 pounds
D (applicable to SSE)	119,000 pounds	226,000 pounds

The platform for Rack B features 1-inch thick vertical plates (ribs) located beneath each of the four rack shear pad bearing surfaces (pedestal locations), which were evaluated for the maximum critical lateral loading stress and bearing stress. The 5/16-inch fillet welds joining the vertical 1-inch thick plates (ribs) to the top 1-inch thick plate are loaded in compression by the vertical bearing loads, so, only the lateral loadings were considered in the evaluation of the welds. The approximate resultant safety factors for the Rack B platform are summarized in the following table.

Platform for Rack "B"		Safety Factor	
		Level A (OBE)	Level D (SSE)
Platform	Lateral	198	164
	Bearing	51	41
Welds	Lateral	4.5	6.75

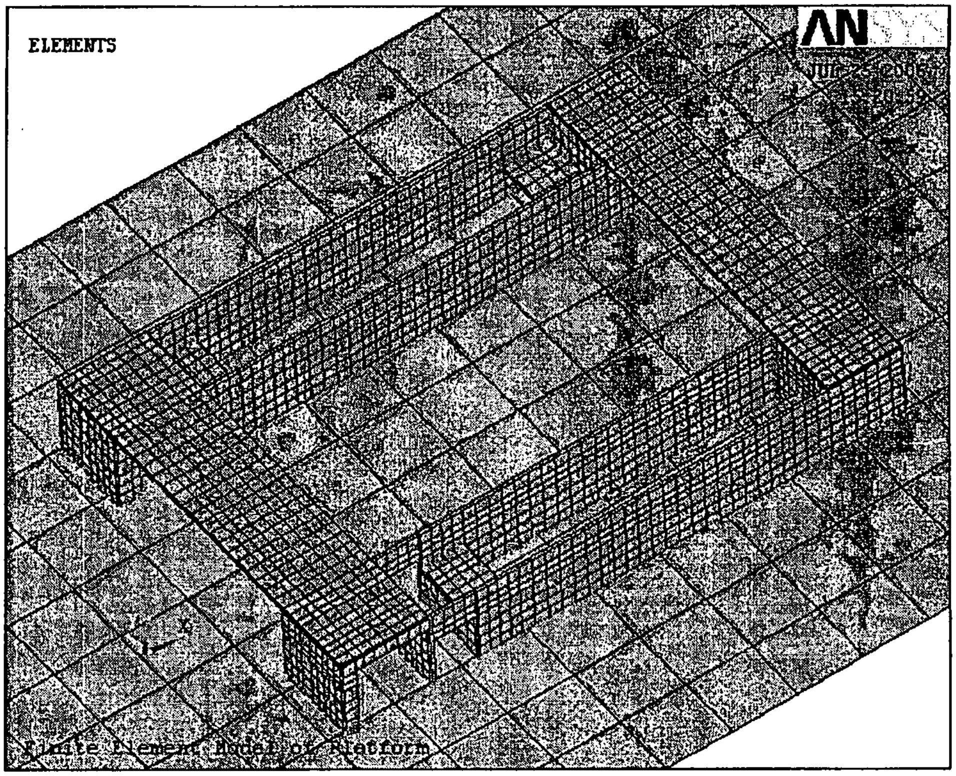


Figure 2
Rack A Platform Finite Element Model

4. NRC Request

Section 8.4 of the Holtec report documents that a modal analysis was performed on the SFP floor, and: "it was determined that the first mode frequency in the vertical direction is 35.4 Hz, which is larger than 33 Hz and, thus, in the rigid range. Therefore, seismic amplification due to slab flexibility is not an issue. These results justify development of equivalent static loads using the zero period accelerations as multipliers".

(a) Table 8.3.1 of the Holtec report indicates that an uncracked section modulus was used for this computation. Please discuss the appropriateness of using an uncracked section modulus to calculate the fundamental frequency of the SFP floor.

(b) How does the calculation for the fundamental frequency of the SFP floor consider the static mass of the SFP water? This load does not appear to be listed in Table 8.5.1 of the Holtec Report. Section 2.3.3.2.4 of the Cooper Nuclear Station Updated Safety Analysis Report notes that the SFP contains about 2.1 million pounds (2,100 kips) of water.

NPPD Response

(a) The use of an uncracked section modulus to calculate the fundamental frequency of the SFP floor is consistent with the chosen method of analyzing the reinforced concrete SFP floor. In this method the mechanical and thermal loads are analyzed separately using uncracked and cracked section properties, respectively, and then the stress results are combined to evaluate the design load combinations. Since the modal analysis of the SFP slab is performed to establish a dynamic amplification factor for computing earthquake loads (i.e., mechanical loads), the use of uncracked section properties is appropriate. Cracked section properties are used only to evaluate thermal loads and to provide a realistic assessment of the redistributed internal forces and moments, as permitted by Section A.3.3 of American Concrete Institute (ACI) 349. The intent of the ACI Committee is further clarified in ACI 349R-85 (Commentary on Code Requirements for Nuclear Safety Related Concrete Structures), which states that the analysis may "consider the structure uncracked for mechanical loads and only consider the effect of cracking on thermal loads." Holtec has used this method of analysis numerous times to qualify reinforced concrete SFP structures, based on an established history of acceptance by the NRC.

(b) The calculated first mode frequency of 35.4Hz for the SFP slab, reported in Holtec Report No. HI-2043224, is based on a 64-inch thick concrete slab ($\gamma = 150$ lb/ft^3) with simply supported boundary conditions and no additional fluid mass. While it is clearly conservative to assume simply supported boundary conditions, it is non-conservative to assume that none of the contained SFP water mass participates in the dynamic response of the SFP slab. To provide a more accurate estimate of the SFP floor fundamental frequency, a series of modal analyses have been performed assuming both clamped and simply supported boundary conditions and increased slab densities to account for half or all of the contained SFP water mass. The following table summarizes the frequency results.

SFP Floor Fundamental Frequency (Hz)

Water mass added	Simply supported edges	Clamped edges
Half	23.1	44.1
Total	18.4	35.1

The minimum result is 18.4 Hz, which represents a conservative lower bound estimate of the slab fundamental frequency since it assumes both simply supported boundary conditions and full participation of the SFP water mass. In reality, the SFP slab behaves more like a rectangular plate with clamped edges, and the mass participation of the SFP water is less than 100% since the water is not rigidly attached to the slab. Therefore, it is reasonable to conclude that the fundamental frequency of the slab is above 20 Hz. Since the vertical SSE response spectrum for the SFP floor, which is shown in Figure 3, has a constant acceleration above 20 Hz, the use of the zero period acceleration (ZPA) to compute the seismic amplification of the SFP slab and the contained SFP water mass is justified, and the minimum safety factors reported in Holtec Report No. HI-2043224 are indeed valid.

Finally, the static mass of the SFP water was inadvertently omitted from Table 8.5.1 of Holtec Report No. HI-2043224. The finite element analysis of the SFP slab conservatively considers a uniform acting pressure of 16.9 psi over the entire SFP slab area. This represents a total hydrostatic load of 2.7 million pounds (2,700 kips), which is significantly more than the contained water mass of 2,100 kips reported in CNS Updated Safety Analysis Report (USAR) Section XII-2.3.3.2.4. For the earthquake load, the hydrostatic load (2,700 kips) is amplified by the vertical ZPA values for OBE (0.0685g) and SSE (0.137g).

5. NRC Request

Please discuss any nonconformances that are related to material degradation issues (concrete, rebar) in the SFP.

NPPD Response

No nonconformances related to material degradation issues in the concrete/rebar structural elements of the CNS SFP have been documented to date. No leakage from the CNS SFP has been identified to date. However, there were two significant nonconformances (events), not related to material degradation issues, which are relevant to the integrity of the CNS SFP. These events involved dropping a core shroud head bolt and dropping a control rod blade in the

SFP. Neither of these two events resulted in any discernible damage to the 1/4-inch thick stainless steel liner plate. The core shroud head bolt did not come into contact with the liner plate. The area of contact/impact of the control rod blade with the liner plate was inspected through the use of an underwater camera. No damage was visible.

6. NRC Request

Table 6.7.1 of the Holtec report specifies 4 percent damping for Operating Basis Earthquake (OBE) and 5 percent damping for Safe Shutdown Earthquake (SSE) as input data for the rack seismic analysis. Table 1 of NRC Regulatory Guide 1.61 specifies 2 percent damping for OBE and 4 percent damping for SSE for welded steel structures. Please discuss the appropriateness of the damping values used in the analysis.

NPPD Response

The CNS design and licensing basis information found in USAR Section XII-2.3.5.2.5 indicates that "*Steel Frame Structures*" are to be analyzed using a damping value of 2.0 percent and "*Welded Assemblies*" are to be analyzed using a damping value of 1.0 percent when conducting dynamic analyses using seismic response spectra methodology. The selection of the CNS design and licensing basis ground response spectra for the seismic design analyses of safety-related Structures, Systems, and Components (SSCs) was completed prior to the October 1973 issuance of NRC Regulatory Guides 1.60, "*Design Response Spectra for Seismic Design of Nuclear Power Plants,*" and 1.61, "*Damping Values for Seismic Design of Nuclear Power Plants.*" The CNS-specific ground response spectra does not completely envelope the Regulatory Guide 1.60 spectra, which precludes the direct use of the higher (less conservative) damping values permitted by Regulatory Guide 1.61 for the analysis of welded steel structures.

The subject CNS OBE floor response spectra (at 4 percent damping) and SSE floor response spectra (at 5 percent damping), were not directly utilized to conduct dynamic response spectra-type analyses of the proposed new storage racks. The subject floor response spectra for the 976'-0" elevation of the Reactor Building were utilized to create artificial acceleration time histories of the dynamic input motion applicable to the location of the proposed fuel storage racks. The synthetic "conversion" of the subject floor response spectra information to artificial time-history input motion was confirmed to be accurate by ensuring that "output" floor response spectra, created from these artificial time-history input motions, would adequately and appropriately envelope the CNS OBE and SSE floor response spectra originally provided to the analysts. These "verified" artificial time histories were then used as input data to conduct the non-linear dynamic analyses of the new storage racks, which are base-supported on the storage pool floor, elevation 962'-3."

Time-history dynamic input motion information is not dependent on an assumed damping level in the structure being dynamically loaded, as the "input" information is in the format of acceleration versus time, rather than a format of acceleration versus structural response (frequency or period). As such, the damping level of the "input" floor response spectra would not be critical to the dynamic analyses of the proposed new storage racks.

The numeric values of the structural damping values assumed in the rack structures were confirmed to be as listed in Table 6.7.1 of the Holtec report (4 percent for the OBE dynamic analyses, and 5 percent for the SSE dynamic analyses). These internal damping values are not in accordance with the CNS USAR, nor are they consistent with Regulatory Guide 1.61. The appropriate structural damping value for use in conducting each of the dynamic analyses for CNS is 1 percent. As such, the assumed structural damping values utilized in the rack dynamic analyses are potentially non-conservative.

As the lowest fundamental mode of horizontal structural response in the proposed new storage racks was determined by analysis to be approximately 7 Hz, and because the input dynamic response was applicable to an elevation higher than the pool floor (976'-0" versus 962'-3"), the effect of this non-conservative assumption is not significant.

The potential increase in seismic response is estimated as follows:

Seismic Level	7 Hz Response at 1% Damping, 958'-3"/ 976'-0" Level	7 Hz Response at 4% Damping, 976'-0" Level	7 Hz Response at 5% Damping, 976'-0" Level	Potential Impact of 1% Damping on Response
OBE	0.37g	0.37g	N/A	Nil
SSE	0.60g	N/A	0.58g	0.02 g (3.3%) increase

The use of potentially non-conservative 4% damping in the rack structure for the OBE analyses has a negligible impact on the response when compared to the required 1% damping response. The use of potentially non-conservative 5% damping in the rack structure for the SSE analyses has a small impact (less than 3.5%) when compared to the required 1% damping response. This small difference is not considered to be significant.

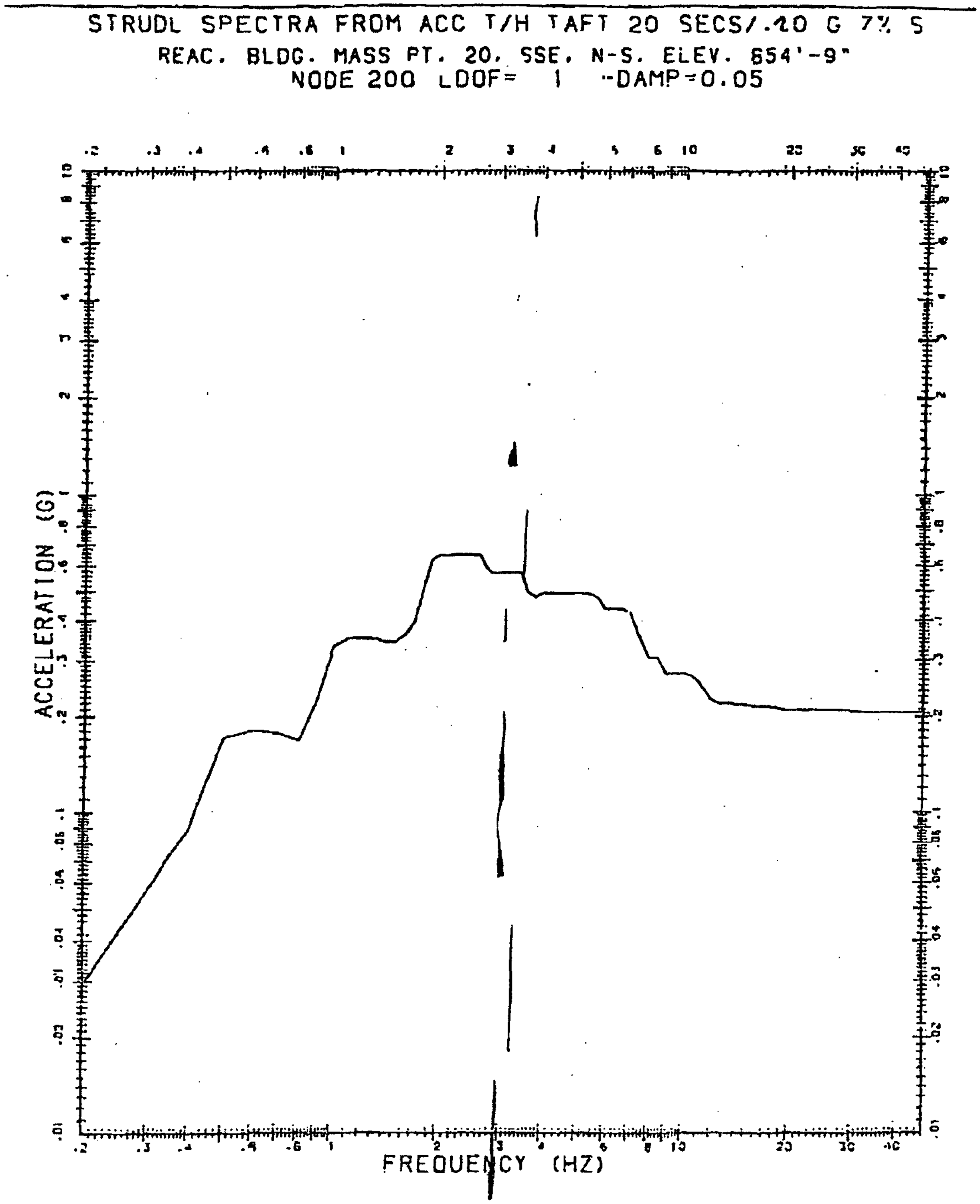
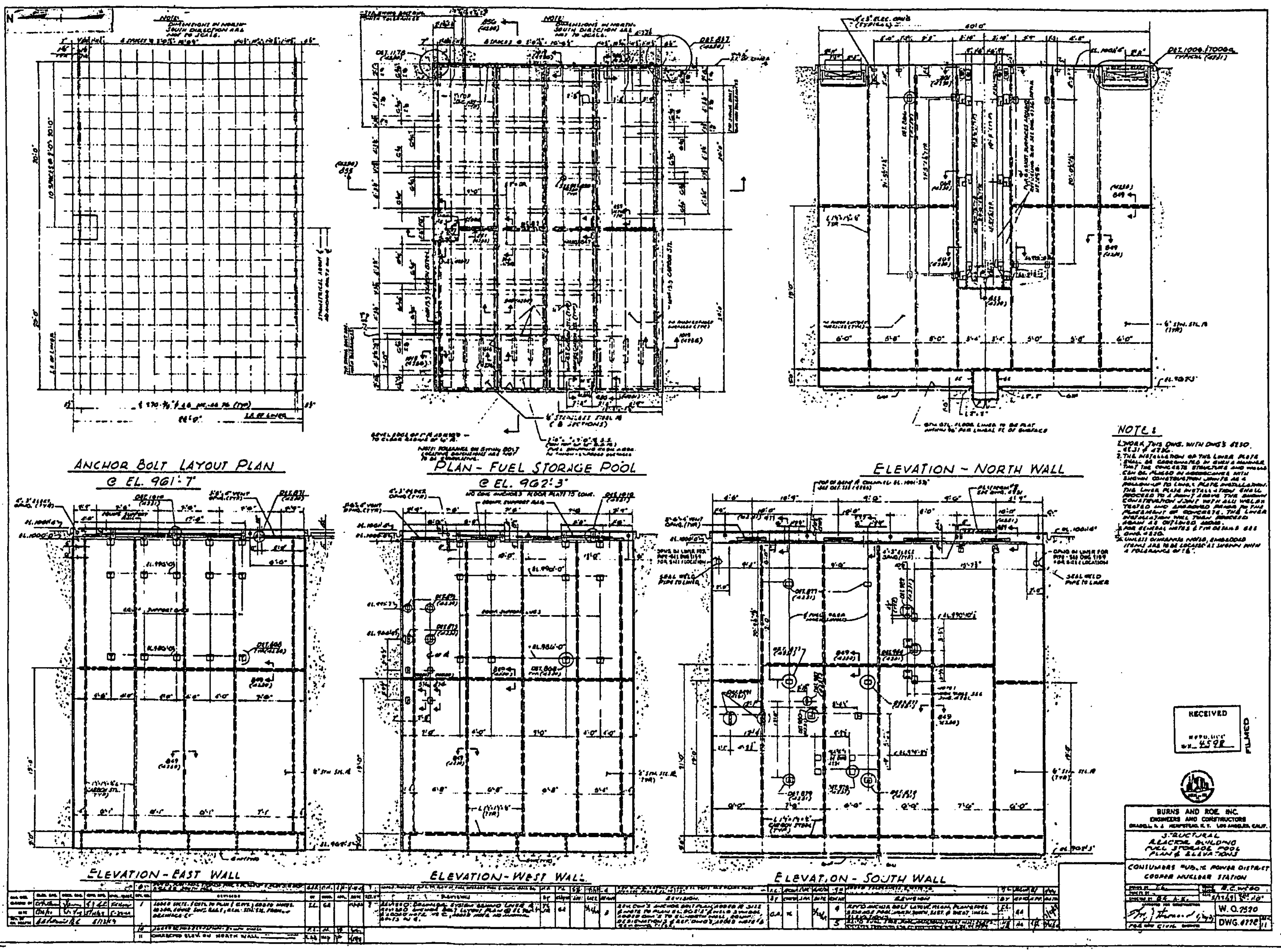


Figure 3
SSE Response Spectra for Vertical Ground Motion
(Note: Vertical ground motion is derived by applying a factor of 2/3 to the above spectrum)

Burns and Roe, Inc.

Drawing Number 4228, Revision 11



ANCHOR BOLT LAYOUT PLAN
@ EL. 961'-7"
PLAN - FUEL STORAGE POOL
@ EL. 962'-3"
ELEVATION - NORTH WALL
NOTES
1. WORK THIS DWG. WITH DWG'S 4230, 4231 & 4736.
ELEVATION - EAST WALL
ELEVATION - WEST WALL
ELEVATION - SOUTH WALL
RECEIVED
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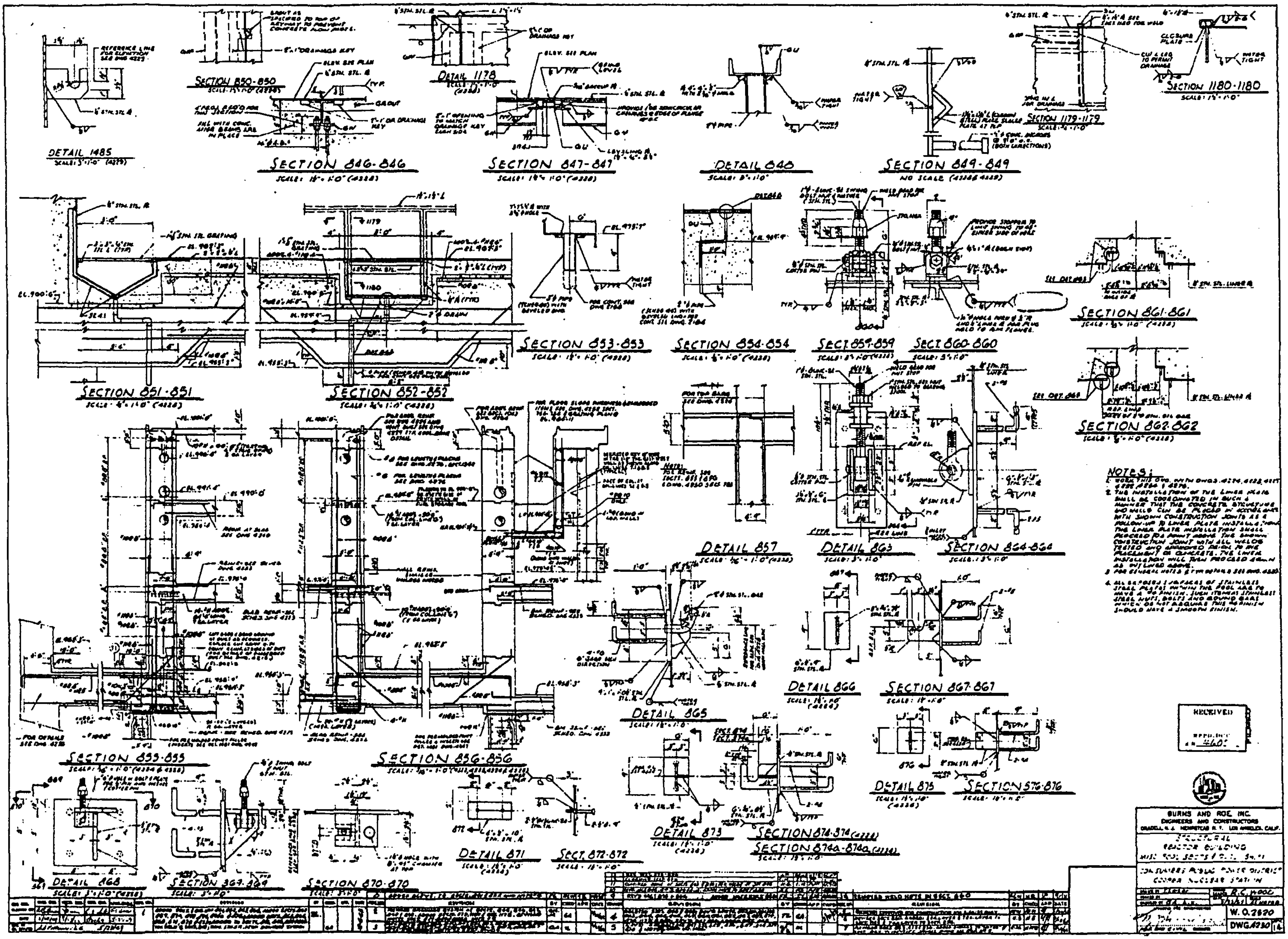
ENGINEERS AND CONSTRUCTORS
STRUCTURAL
REACTOR BUILDING
FUEL STORAGE POOL
PLAN & ELEVATIONS
CONSUMERS PUBLIC POWER DISTRICT
COOPER NUCLEAR STATION
W. O. 7520
DWG. 4778

Burns and Roe, Inc.

Drawing Number 4230, Revision 14



DETAIL 1485
SECTION 850-850
SECTION 846-846
DETAIL 1178
SECTION 847-847
DETAIL 848
SECTION 849-849
SECTION 1179-1179
SECTION 1180-1180
SECTION 851-851
SECTION 852-852
SECTION 853-853
SECTION 854-854
SECT 859-859
SECT 860-860
SECTION 861-861
SECTION 862-862
NOTES:
DETAIL 857
DETAIL 863
SECTION 864-864
DETAIL 866
SECTION 867-867
DETAIL 865
SECTION 855-855
SECTION 856-856
DETAIL 875
SECTION 876-876
DETAIL 873
SECTION 874-874
SECTION 874a-874a
DETAIL 871
SECT. 872-872
DETAIL 868
SECTION 869-869
SECTION 870-870
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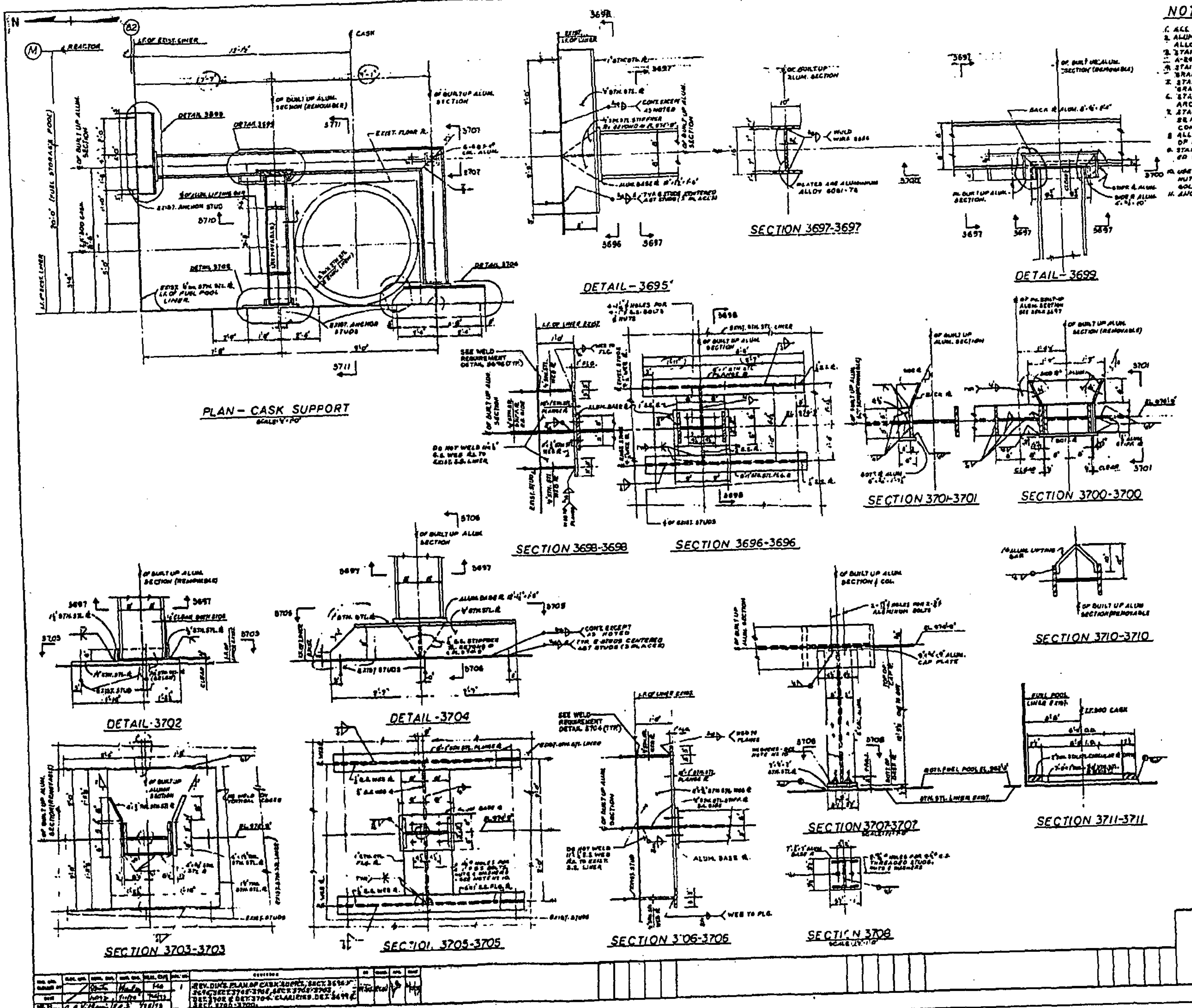
R.C. WOOD
W.O. 2820
DWG 4230
14

Burns and Roe, Inc.

Drawing Number 4288, Revision 1



NOTES:

1. ALL ALUMINUM TO BE ALLOY 6061-T6.
2. ALUMINUM WELDING MATERIAL TO BE FILLER ALLOY 5356.
3. STAINLESS STEEL PLATES TO CONFORM TO ASTM A-240, TYPE 304
4. STAINLESS STEEL BOLTS TO CONFORM TO ASTM. A193 GRADE B8.
5. STAINLESS STEEL NUTS TO CONFORM TO ASTM A194 GRADE B8.
6. STAINLESS STEEL WELDING TO BE SHIELDED METAL ARC USING ELECTRODE SFA-5.4-E 308.
7. STAINLESS STEEL THREADED STUD AND NUT SHALL BE AS MANUFACTURED BY THE NELSON STUD WELDING COMPANY OR APPROVED EQUAL.
8. ALL WELDING TO CONFORM TO THE RECOMMENDATIONS OF A W S CODE.
9. STAINLESS STEEL PL. SHALL BE HOT ROLLED, ANNEALED & PICKLED, IN ACCORDANCE WITH ASTM A-480
10. USE STAINLESS STEEL CIRCULAR WASHERS UNDER NUTS OR BOLT HEADS ON THE ALUMINUM SIDE OF BOLTED CONNECTIONS.
11. ANODIZE ALL ALUMINUM AFTER FABRICATION.

RECEIVED

BURNS AND ROE, INC.
ENGINEERS AND CONSTRUCTORS
ORADELL, N.J. HEMPSTEAD, N.Y. LOS ANGELES, CALIF.

STRUCTURAL
REACTOR BUILDING
I.F. 300 CASK SUPPORT-PLAN, SECT. & DETL.

CONSUMERS PUBLIC POWER DISTRICT
COOPER NUCLEAR STATION

W. O. 2520

DWG. 4268

Holtec Drawing Number 4732

Rack A Support Platform

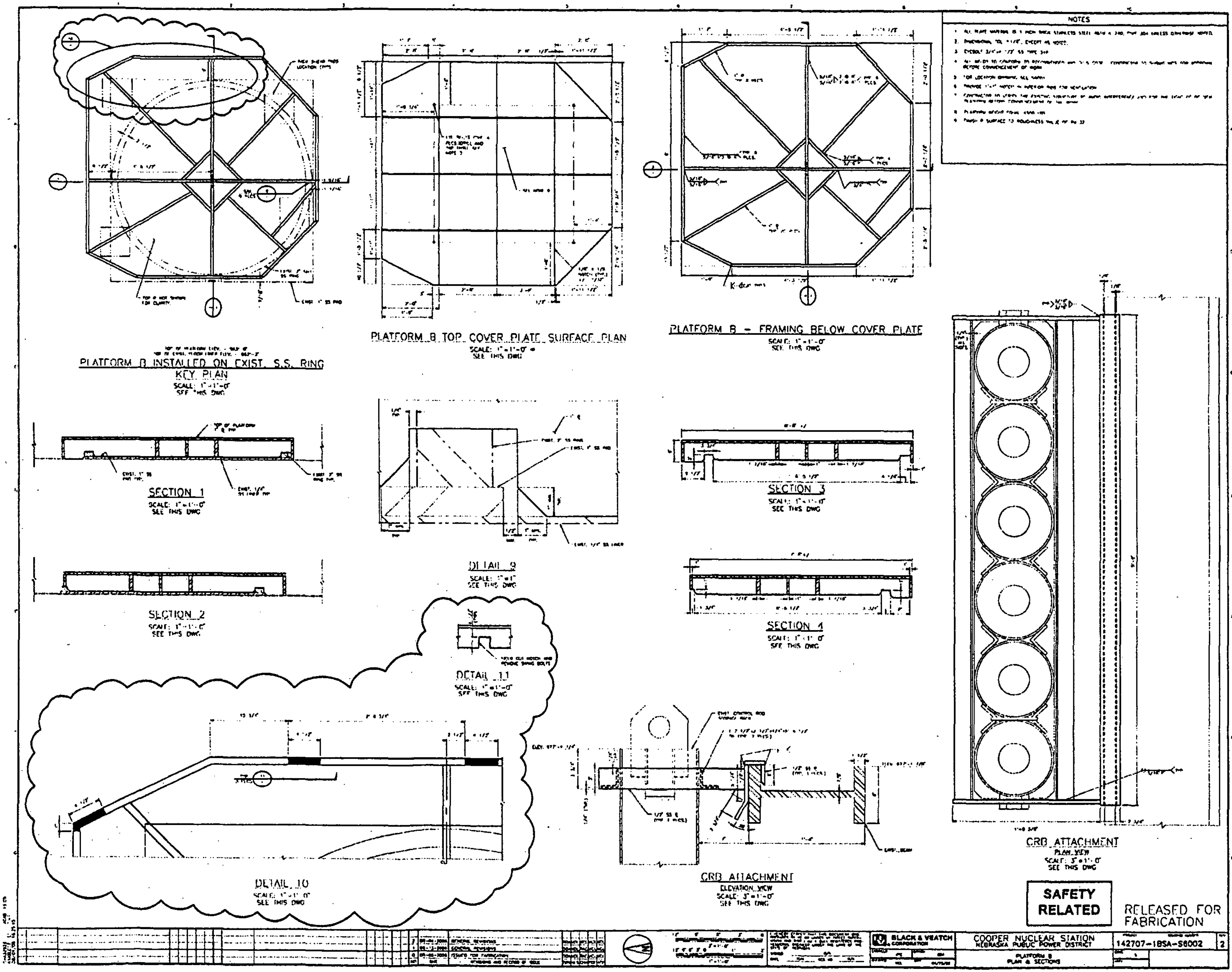
(5 sheets)

This page is provided for use in place of Page 19 of 26 from Attachment 2 of NLS2007012, the cover page for the proprietary five-page Holtec Drawing 4732, as well as the five pages of the proprietary Holtec drawing (page 20 of 26 through page 24 of 26).

This page does not contain proprietary information and is suitable for use in the copy of NLS2007012 that is provided for public disclosure.

Black and Veatch Corporation Drawing No. S6002.

Platform B – Plan & Sections


NOTES
PLATFORM B INSTALLED ON EXIST. S.S. RING
KEY PLAN
SCALE: 1"=1'-0"
SEE THIS DWG
PLATFORM B TOP COVER PLATE SURFACE PLAN
SCALE: 1"=1'-0"
SEE THIS DWG
PLATFORM B - FRAMING BELOW COVER PLATE
SCALE: 1"=1'-0"
SEE THIS DWG
SECTION 1
SCALE: 1"=1'-0"
SEE THIS DWG
DETAIL 9
SCALE: 1"=1'
SEE THIS DWG
SECTION 3
SCALE: 1"=1'-0"
SEE THIS DWG
SECTION 2
SCALE: 1"=1'-0"
SEE THIS DWG
SECTION 4
SCALE: 1"=1'-0"
SEE THIS DWG
DETAIL 11
SCALE: 1"=1'-0"
SEE THIS DWG
DETAIL 10
SCALE: 1"=1'-0"
SEE THIS DWG
CRB ATTACHMENT
ELEVATION VIEW
SCALE: 3"=1'-0"
SEE THIS DWG
CRB ATTACHMENT
PLAN VIEW
SCALE: 3"=1'-0"
SEE THIS DWG
SAFETY RELATED
RELEASED FOR FABRICATION
BLACK & VEATCH CORPORATION
COOPER NUCLEAR STATION
NEBRASKA PUBLIC POWER DISTRICT
PLATFORM B
PLAN & SECTIONS
142707-1BSA-S6002
2

Attachment 3

Response to Steam Generator Tube Integrity and Chemical Engineering Branch Request for Additional Information Regarding License Amendment Request for Spent Fuel Pool Storage Expansion

Cooper Nuclear Station, Docket No. 50-298, DPR-46

Background

The proposed change revises TS 4.3, "Fuel Storage," to reflect the addition of fuel storage capacity in the spent fuel pool by adding up to two additional storage racks. The increased fuel capacity involves use of the Metamic® neutron absorbing material.

The staff has conditioned prior acceptance of the Metamic® material for use in spent fuel pools upon a coupon sampling program (e.g., Clinton, Arkansas Nuclear One) to ensure performance consistent with that described in the engineering reports.

An acceptable coupon sampling program has the following attributes:

- *The size and types of coupons are similar in fabrication and layout to the proposed application, including welds or mechanical fasteners and proximity to stainless steel.*
- *The technique for measuring the initial B_4C content of the coupons is described.*
- *Scratches and other surface anomalies on the coupons simulate as-installed conditions of Metamic® panels.*
- *Justification of the frequency of coupon sampling is provided.*
- *Tests to be performed on coupons are specified; e.g., weight measurement, measurement of dimensions (length, width and thickness), and B_4C content and acceptance criteria.*

1. NRC Request

   Provide a detailed description of your coupon sampling program.

   NPPD Response

   A detailed description of the coupon sampling program is provided in Attachment 4.

2. NRC Request

Please describe the scope of the receipt inspections performed on the Metamic® panels to establish a baseline. In addition, discuss who will perform these receipt inspections and where they will be performed.

NPPD Response

Metamic, LLC, manufactures the Metamic panels that are used in the spent fuel racks, and ships them to Holtec's manufacturing facility, where the panels are incorporated into the spent fuel storage racks. Prior to shipping the panels to the Holtec manufacturing facility, Metamic, LLC, performs the following inspections of the panels:

- A dimensional inspection of the width, length, and gauge (thickness) on 100 percent of the final panels.
- A visual inspection on 100 percent of the Metamic panels.
- A check of the areal density on 10 percent of the panels using neutron attenuation measurements to confirm conformance to minimum Boron-10 content specification requirements.

Metamic, LLC, also provides the chemical certification for the boron carbide (B_4C) in each batch that is used in the fabrication of the panels to Holtec.

Holtec then independently inspects a sample of the Metamic panels for correct width, length, and gauge prior to use of the panels in manufacturing the fuel racks.

3. NRC Request

Please discuss the surface preparation or anodizing process that will be used on the Metamic® panels. In addition, what cleaning process will be used to ensure sufficient removal of surface contaminants? Please note: the surface preparation for the coupons is to be similar to what is used on the Metamic® panels.

NPPD Response

The Metamic® production stock (the material from which the panels are extruded) is mechanically cleaned using an automated air-blast system with alumina oxide abrasive prior to forming the material into the panels. This eliminates any potential surface contamination that might occur during the extrusion process. After the panels are fabricated to final gauge they are cleaned with a nitric acid/water solution to neutralize and passivate the surface of the final product. This is the standard surface preparation used on all Metamic® panels and retained samples of Metamic® panels.

4. NRC Request

Please discuss how the Metamic® panels are to be attached to the new spent fuel racks (welding, mechanical fasteners, etc). If the Metamic® panels are to be welded, what welding process will be used and what weld filler material will be used?

NPPD Response

The Metamic® panels are not directly welded to the spent fuel racks. Rather, the Metamic® panels are placed into a stainless steel pocket, and this pocket, containing the Metamic®, is welded to the walls of the rack cells.

5. NRC Request

Given that Metamic® is a new material for SFP applications, we request that you submit the results of each surveillance to the staff. The reporting requirements should be addressed in your surveillance program. The reporting requirements should include the baseline inspection results and the results of all subsequent inspections (i.e., visual, length, thickness, weight, and neutron absorption).

NPPD Response

NPPD will obtain the baseline data on the coupons prior to placing the rack and the coupon tree into the SFP and submit the results to the NRC. NPPD will also remove and test the coupons on the schedule provided in the description of the coupon sampling program, provided in Attachment 4, and will submit the result of the subsequent coupon surveillance and testing to the NRC. The description of the program states that the program will require submittal of the baseline data and the results of each coupon surveillance to the NRC.

Neutron attenuation testing must be performed by an outside laboratory. As a result, it is estimated that six months following removal of the coupons from the SFP will be needed to obtain data and test results, and to submit the test results to the NRC.

Attachment 4

Coupon Sampling Program for Metamic-Poisoned Spent Fuel Racks

Cooper Nuclear Station, Docket No. 50-298, DPR-46

Purpose/Objective

The purpose of the Metamic coupon sampling program is to characterize certain physical and chemical properties of the Metamic sample coupon from the CNS spent fuel storage pool. The primary objective is to provide data necessary to confirm the capability of the poison material in the racks to continue to maintain K_{eff} within limits.

Program Requirements

The coupon samples will contain 25% boron carbide, which is consistent with the boron carbide content used in the new CNS spent fuel storage racks. Each coupon is nominally 6 inches long, 4 inches wide and 0.075 inches thick. Each coupon will be mounted on the coupon tree. The coupon tree holds 8 coupons.

The CNS Metamic coupon sampling program will be incorporated into CNS station procedures. The coupon tree will be placed in the spent fuel pool at a location that will ensure a representative gamma dose to the coupons. The program will require a coupon to be removed from the spent fuel pool for testing after 2, 4, 8, 12, 16, 20, 24, and 28 years. If no failures of test acceptance criteria are identified after the first 3 coupons are pulled and tested, the test intervals may be evaluated for possible extension. After testing, coupons will be returned to the coupon tree to support long term testing as required. If additional testing is required after the original 8 samples have been tested once, sample testing will begin again with the first sample withdrawn.

The coupon sampling program is intended to monitor the change in physical properties of the absorber material. Prior to initial installation in the spent fuel pool environment, each coupon is pre-characterized. The physical characteristics presented below are documented for each coupon and included in the documentation package for the coupon tree. The coupons, when initially measured, are in a post-manufactured state and have not been irradiated or exposed to a spent fuel pool environment. These measurements define the baseline for the coupons.

Upon removal from the spent fuel pool in accordance with the sampling program, the measurements to be performed included the following:

- Visual observation and photography
  - o Observe for visual indications such as bubbling, blistering, cracking, or flaking

  - Photograph both sides of exposed coupon to document coupon condition
- Dimensional measurements
  - Length
  - Width
  - Thickness
- Mass

The results from the above measurements and physical observations will be recorded and evaluated for any physical or visual change relative to the pre-characterized data. In the event there is a physical (i.e., measurement) or visual change outside of the following allowable tolerances, then the program will dictate that neutron attenuation testing be performed on the coupon to confirm neutron attenuation capabilities:

- Length and Width: +/- 0.125 inches
- Thickness: +/-0.07 inches
- Mass: +/-5%

In addition to the above measurements and observations of the coupon physical characteristics, a neutron attenuation test will be performed on a coupon sample after 4, 12, and 20 years to confirm neutron attenuation capabilities of the Metamic material. This testing will be performed regardless of whether the physical measurements taken at these intervals are within the allowable tolerances or not.

The program will require submittal of the baseline data on unirradiated coupons and the results of each coupon surveillance to the Nuclear Regulatory Commission.

Summary

This sampling program will regularly monitor the Metamic coupons' critical parameters, which have historically indicated degradation in other neutron absorbing materials. Coupon changes outside of the prescribed physical acceptance criteria will result in additional testing activities that will directly assess the neutron attenuation capabilities of the Metamic compound. In addition, periodic testing of the neutron attenuation capabilities of the coupon material will provide a correlation between the neutron absorption capabilities and the physical characteristics of the coupons.

NPPD has concluded that this sampling program provides an efficient and effective means to monitor the condition of the Metamic compound and to detect any potential degradation that could lead to a potential change in neutron absorption capability.

0.ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2007012

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
NPPD will develop a procedure implementing the coupon sampling program, as discussed in Attachment 4 of NLS2007012.	NLS2007012-01	Prior to installation of the Metamic-poisoned spent fuel storage rack.
NPPD will obtain baseline data taken on the unirradiated Metamic coupons and submit that data to the NRC.	NLS2007012-02	Prior to installing the coupon tree with the Metamic coupons.
NPPD will remove a coupon and perform testing and surveillance on the coupon after 2, 4, 8, 12, 16, 20, 24, and 28 years following initial placement of irradiated fuel in the SFP, and will submit the results the NRC.	NLS2007012-03	Beginning with Operating Cycle 25 (approximately May 2008), after the following periods: • 2 years + 6 months • 4 years + 6 months • 8 years + 6 months • 12 years + 6 months • 16 years + 6 months • 20 years + 6 months • 24 years + 6 months • 28 years + 6 months
(Revised commitment from NLS2006028): Two rows of 5-year cooled fuel will be placed along the sides of the new racks facing the fuel pool walls to provide shielding from freshly discharged fuel assemblies. The procedure for controlling storage of spent fuel in the spent fuel pool will be revised to require the placement of two rows of 5-year cooled fuel.	NLS2006028-01	Placement of the new racks in the spent fuel pool