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PNP 2012-071

September 6, 2012

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

SUBJECT: Response to Request for Additional Information – License Amendment Request – Replacement of Spent Fuel Pool Region I Storage Racks

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

References:
1. ENO letter "License Amendment Request – Replacement of Spent Fuel Pool Region I Storage Racks" dated February 28, 2012 (ADAMS Accession Numbers ML12061A288, ML12061A289, and ML12061A290)
2. NRC electronic mail of July 19, 2012, "ME8074_RAI"
3. NRC electronic mail of July 24, 2012, "Request for Additional Information (Revised) – Palisades – SFP Rerack LAR- ME8074"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request (Reference 1) to modify the Renewed Facility Operating License, Appendix A, Technical Specifications (TS), as they apply to the replacement of the spent fuel pool region I storage racks. ENO received an electronic request for additional information in Reference 2, which was later revised in Reference 3.

Attachment 1 provides the ENO response to the RAI. Two revised clean and marked-up TS pages are included in Attachments 2 and 3 respectively.

~~**PROPRIETARY**~~
Withhold from public disclosure under 10 CFR 2.390. When separated from Attachment 5 the remainder of the submittal may be decontrolled.

Attachment 4 provides the Holtec International proprietary authorization affidavit supporting the proprietary nature of Attachment 5. The affidavit sets forth the basis for which the information may be withheld from public disclosure by the NRC and addresses the specific considerations listed in 10 CFR 2.390, *Public inspections, exemptions, request for withholding.*

Attachment 5 contains proprietary Holtec International information to respond to NRC Reactor Systems (SRXB) RAI item 4.

A copy of this response, without the proprietary information, has been provided to the designated representative of the State of Michigan.

This letter contains no new or revised commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 6, 2012.

Sincerely,

ajv/jlk

Attachments:
1. Response to Request for Additional Information Regarding License Amendment Request for Replacement of Spent Fuel Pool Region I Storage Racks
2. Revised Technical Specifications Pages 4.0-4 and 4.0-5
3. Mark-up of Technical Specifications Pages 4.0-4 and 4.0-5
4. Holtec International Affidavit
5. Holtec International Proprietary Information – Response to Reactor Systems (SRXB) Item 4

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

ATTACHMENT 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT REQUEST
FOR REPLACEMENT OF SPENT FUEL POOL REGION 1 STORAGE RACKS

By letter dated February 28, 2012 (ML12061A288, ML12061A289, and ML12061A290), Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request (LAR) to revise Appendix A, Technical Specifications (TS), as they apply to the Palisades Nuclear Plant (PNP) spent fuel pool (SFP) storage requirements in TS Section 3.7.16 and criticality requirements for Region I SFP and north tilt pit fuel storage racks, in TS section 4.3.

A request for information was received from the Nuclear Regulatory Commission (NRC) by electronic mail on July 19, 2012, and amended on July 24, 2012. Provided below is the ENO response to the RAI.

<u>Reactor Systems (SRXB)</u>
NRC Request

1. *Appendix D, "Guidance on Spent Fuel Pool Racks" of Standard Review Plan (SRP) Section 3.8.4, "Other Seismic Category I Structures," Revision 3, stated that "Because of gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads resulting from this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to damage of the fuel." Please provide fuel handling accident analysis and any associated information regarding the damage fuel during the postulated "fuel to fuel" drop event, and the associated decontamination factor, compared to the design-basis FHA analysis.*

ENO Response

1. The consequent loads on the stored fuel assemblies due to seismic impacts are discussed in the response to item 2 from the Mechanical and Civil Engineering Branch below. In short, the maximum calculated impact load on a spent fuel assembly (2.21-g) is well below the failure limit of a PWR 15 x 15 fuel assembly (22-g), which is the same fuel type at PNP, as determined in Appendix C to NUREG-1864, *A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant*. Thus, the design basis earthquake at PNP will not cause any fuel damage. The fuel handling accident analyses performed to

demonstrate the continued functional capability of the SFP racks are summarized in Chapter 7 of Attachments 5 and 6 to the LAR, and they are further discussed in the responses below to item 7 and item 8 from the Mechanical and Civil Engineering Branch. No "fuel-to-fuel" drop event was analyzed for this LAR since the consequences of such an event are not affected by the SFP racks.

NRC Request

2. In Section 4.2.3.1, the licensee states that the design basis assembly is determined with a single cell MCNP5-1.51 Region I model. Please provide more detailed analysis and sensitivity studies on how the design basis assembly is appropriate for the specific condition of the Region I pool.

ENO Response

2. The LAR discusses the determination of the design basis analysis in Section 4.2.3.1, Section 4.5.1 and Section 4.6.1.1. As discussed in those sections, the full inventory of fuel designs was reviewed and it was determined that all the fuel designs could be separated into two categories based on the active fuel length, cladding inside diameter (ID) and pellet diameter. All other parameters relevant to this criticality analysis were equivalent or dispositioned (see the discussion in Section 4.5.1). The two categories were based on the fuel design batch orders (age) and are described by those fuel assemblies from before Batch R and those in Batch R and beyond. Specific studies were performed for these two fuel designs and the bounding fuel design was selected for use as the design basis fuel assembly. These studies are discussed in Section 4.6.1.1 and presented in Table 4.6.1 for both pure and borated water. All further design basis and accident calculations use the bounding fuel assembly. Furthermore, the studies used to determine the design basis fuel assembly used the same single cell MCNP5-1.51 model with reflective boundary conditions as all of the other studies and calculations performed to determine keff at the 95/95 level (i.e. tolerance and uncertainty calculations). Therefore, the design basis fuel assembly calculations are appropriate to the specific condition of the Region 1 racks in the SFP and no additional studies or detailed analysis is required.

NRC Request

3. In Section 4.2.3.5, the licensee states that "any missing fuel rod [from a fuel rod assembly] replaced with a solid material is acceptable." The licensee then provided two cases, "Missing Four Fuel Rods from One Assembly in a 3x3 Array" and "Missing Four Fuel Rods from Every Assembly in SFP." What replacement materials were analyzed that are considered acceptable? Also, please provide analysis that the two scenarios analyzed in Section 4.2.3.5 are the most limiting cases.

ENO Response

3. The discussion in Section 4.2.3.5 is related to both the replacement of a missing fuel rod with a solid rod and to fuel assemblies with missing rods that are not replaced.

With respect to missing fuel rods that are replaced with a solid rod, the replacement material type has always been, and would be, administratively controlled in the future to be stainless steel or Zircaloy. These two materials are non-fissile and also do not moderate neutrons. Thus, replacing a fuel rod with either a stainless steel or Zircaloy rod of equivalent or greater diameter fuel rod will always result in a less reactive fuel assembly than one with all fuel rod locations.

With respect to the discussion in Section 4.2.3.5 related to fuel assemblies with missing rods that are not replaced, and the two sets of associated calculations (Scenario 1 and Scenario 2), it is noted that for the existing population of fuel assemblies in the SFP, 34 assemblies with missing fuel rods that have not been replaced with a solid stainless steel or Zircaloy rods, have a maximum enrichment of less than 3.3 weight percent (wt%) U-235. For these low enriched fuel assemblies, engineering judgment is sufficient to qualitatively determine that any potential reactivity effect due to missing fuel rods is well bounded by the Region 1 analysis that uses 5.0 wt% U-235. Therefore, for the existing fuel population, the missing fuel rods are acceptable and nothing further is required to qualify them for storage in the Region 1 racks.

The calculations in the analysis related to Scenario 1 and Scenario 2 were performed to show what the reactivity effect of missing rods is for the bounding fuel assembly at 5.0 wt%U-235. Note that fuel assemblies do not typically have rods removed prior to depletion, and since this application does not credit burnup, the calculations are performed using fresh fuel. However, any reactivity effect on spent fuel would be well below the calculated reactivity to qualify fresh fuel. Due to the large number of possible permutations and the resulting calculational burden, the calculations that were performed were selected based on scoping studies which determined the likely worst case locations. These cases were used as the basis for the presented permutations. Two sets of calculations were performed: Scenario 1) one fuel assembly with four missing rods surrounded by fuel assemblies without missing rods, and Scenario 2) every fuel assembly in the pool had missing rods. The reactivity effect of Scenario 1 is typically within 2-3 sigma of the design basis. Since Scenario 1 is more realistic than Scenario 2, the reactivity effect of Scenario 1 was used for calculating maximum k_{eff}. Furthermore, it was also shown that even if the overly conservative Scenario 2 is considered, the maximum k_{eff} would still be acceptable. The reactivity effect of missing fuel rods is not significant and is typically negative when crediting soluble boron. Additional calculations to justify the selection of the rod locations are, therefore, not necessary.

NRC Request

4. *In Section 4.2.2.1, the licensee stated that actinides and HTC experiments were considered when performing the benchmarking calculations for MCNP5. Please provide the trending analysis that shows the applicability of this data.*

ENO Response

4. The response to the Reactor Systems (SRXB) item 4 request is considered to be proprietary information by Holtec International and is provided as such in Attachment 5.

NRC Request

5. *The licensee did not specify that the B-10 areal density in the Metamic would be incorporated into the Technical Specifications (TS). Please include this information in the TS.*

ENO Response

5. The minimum Metamic B^{10} areal density is specified in the Holtec International Report No. HI-211504, Table 4.5.3(a), in Attachments 5 and 6 of the LAR. The minimum areal density is also described, in the report, as corresponding to the minimum B_4C loading in Metamic, and the minimum poison thickness. As requested, revised clean and marked-up pages to the proposed TS are included in Attachment 2 and 3 of this letter respectively. A new TS 4.3.1.2g. is included in the revised pages, and states:

   "g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2."

   Editorial and punctuation changes or corrections are made to the TS in the revised pages. The "and" is moved from TS 4.3.1.2d., where it was incorrectly placed, to the end of TS 4.3.1.2f. Periods at the end of TS 4.3.1.2e. and TS 4.3.1.2f. were changed to semi-colons. The semi-colon at the end of the second line in TS 4.3.1.3 is changed to a colon.

   The revision to add the minimum B^{10} areal density has affected TS pages 4.0-4 and 4.0-5, which are the only pages included in Attachments 2 and 3 of this RAI response letter. Adding new TS 4.3.1.2g., on page 4.0-4, resulted in moving TS 4.3.1.3e. to the next page, 4.0-5. These two pages, in each of the Attachments 2 and 3, replace those previously submitted in the LAR.

The proposed revision, to add the minimum Metamic B^{10} areal density, along with the editorial and punctuation changes, does not change the intent of the LAR submitted February 28, 2012, and does not affect the no significant hazards analysis in the LAR.

Mechanical and Civil Engineering Branch
NRC Request

1. Section 5.7.6.1, "Region 1 Racks," and Section 5.7.6.2, "Region 2 Racks," of the Palisades Final Safety Analysis Report (FSAR) both indicate that the racks are designed in accordance with Standard Review Plan (SRP or NUREG-0800) Section 3.8.4, "Other Seismic Category I Structures." In Section 6.2.2 of Attachment 6 to the LAR submittal, Revision 2 of SRP Section 3.8.4 is listed as the version used to demonstrate acceptability of the proposed racks for Palisades. However, the latest revision to SRP Section 3.8.4 is Revision 3. Please provide a justification for the use of the dated revision of SRP Section 3.8.4 and reconcile any technical variations between the two revisions as they relate to the design of the replacement spent fuel racks at Palisades.

ENO Response

1. The differences between Revision 2 and Revision 3 of SRP Section 3.8.4 are insignificant and have no effect on the replacement SFP racks at PNP. In Revision 3, of SRP Section 3.8.4, subsection I, item 5, on page 3.8.4-7, is updated to state the following:

> "The organization responsible for quality assurance performs the reviews of design, construction, and operations phase quality assurance programs under SRP Chapter 17. In addition, while conducting regulatory audits in accordance with Office Instruction NRR-LIC-111 or NRO-REG-108, "Regulatory Audits," the technical staff may identify quality-related issues. If this occurs, the technical staff should contact the organization responsible for quality assurance to determine if an inspection should be conducted."

This is the only difference between Revision 2 and Revision 3 of SRP Section 3.8.4. Appendix D to SRP Section 3.8.4, which provides guidance on SFP racks, is wholly unchanged in Revision 3 from Revision 2. Thus, the design and analysis of the replacement SFP racks at PNP is not impacted by these changes. Therefore, the design and analysis of the replacement SFP racks are in compliance with Revision 3 of SRP Section 3.8.4.

NRC Request

2. *Appendix D to SRP Section 3.8.4, Revision 3, "Guidance on Spent Fuel Pool Racks", states that because of gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Subsequently, Appendix D states that it should be demonstrated that the consequent loads on the fuel assembly do not lead to damage of the fuel. Please confirm that the time-history analyses performed for the Palisades SFP rack design have demonstrated that the consequent loads on the fuel assemblies do not lead to damage of the fuel. This confirmation should include information related to the structural capacity of the most limiting spent fuel bundles at Palisades, including material properties, and a demonstrated ability of the bundles to withstand the impact loads generated in the time history analyses.*

ENO Response

2. The time-history analyses performed for the PNP SFP rack design demonstrate that the maximum computed fuel impact loads are well below the failure limit of the fuel bundles. Specifically, in Section 6.7.2 of Attachment 5 & 6 to the LAR, it is demonstrated that the peak fuel-to-cell wall impact load is equivalent to a 2.21-g impact deceleration, which is significantly less than the 60-g design basis deceleration limit that has been approved by the NRC for Holtec's HI-STAR 100 spent fuel transportation cask. In addition, Appendix C to NUREG-1864 evaluates the failure limit of various fuel assembly types due to vertical impact loads, which is the most limiting load direction for spent fuel assemblies due to buckling of the fuel rods. The minimum calculated failure limit, which corresponds to a PWR 15 x 15 fuel assembly (which is the same fuel type at PNP), is 22-g. The above referenced g-load limits are an order of magnitude greater than the maximum computed fuel impact load for the PNP SFP rack design. Therefore, the consequent loads on the fuel assemblies due to a design basis earthquake at PNP will not cause any fuel damage.

NRC Request

3. *In Table 1 of Appendix D to SRP 3.8.4, Revision 3, for Level B Service Limits, under load combination $D + L + T_o + P_f$. P_f is defined as follows:*

   P_f: Force on the racks caused by postulated stuck fuel assembly. This load is considered to be an accident condition.

   Please discuss why the load due to P_f was not included in Section 6.2.2.b of Attachment 6 to the LAR submittal, "Code Stress Limits under Different Service Conditions."

ENO Response

3. The load combination $D + L + T_o + P_f$ was inadvertently omitted from the load combination table presented in Section 6.2.2.b, of Attachments 5 & 6, to the LAR. However, the subject load combination has been analyzed for the PNP SFP rack design, as evident from the discussion on page 7-4, of Attachments 5 & 6, under the heading "Stuck Fuel Event."

   In the analysis, the stuck fuel assembly is conservatively assumed to produce a 5,000 lb vertical force acting on a single rack cell. The analysis demonstrates that the resulting axial stress in the cell and the shear stress in the cell-to-cell welds adjacent base metal are less than the corresponding Level A service limits per ASME Section III, Subsection NF for Class 3 component supports. It is noted that the use of Level A service limits is conservative as SRP Section 3.8.4 specifies Level B service limits for the load combination $D + L + T_o + P_f$. Based on Level A service limits, the minimum computed safety factor for the replacement SFP racks at PNP, due to the postulated stuck fuel assembly, is 1.278.

NRC Request

4. Section 3.8.4 of the SRP and the NRC position paper on spent fuel storage and handling applications indicate that differential thermal expansion loads under normal conditions (T_o) and differential thermal expansion loads under abnormal conditions (T_a) are to be used in combination with primary stresses in loading combinations when determining the structural adequacy of the SFP rack structures. However, Section 6.8.2 of Attachment 6 to the LAR submittal, "Analysis of Thermal Effects," states that thermal stresses do not require evaluation under Subsection NF (of the American Society of Mechanical Engineers Boiler + Pressure Vessel Code). Please provide justification for evaluating the secondary and primary stresses separately in the structural analyses of the cell-to-cell welds for the proposed replacement rack structures. Additionally, please confirm that the guidance of SRP 3.8.4 and the NRC position paper, relative to combining thermal and primary loads, has been considered in the Palisades SFP re-rack analysis and design.

ENO Response

4. As indicated in Section 6.2.2 of Attachment 5 & 6 to the LAR, the load combinations and acceptance limits used for the structural design and qualification of the replacement SFP racks at PNP, are in accordance with SRP Section 3.8.4, which provides the latest NRC staff guidance on SFP racks, and supersedes the load combinations and acceptance limits from the 1979 NRC position paper.

SRP Section 3.8.4 states that the SFP racks can be designed per the requirements of the ASME Code, Section III, Division 1, Subsection NF, for Class 3 component supports. Consequently, subparagraph NF-3121.11 of the ASME Code clearly states:

> "Thermal stress is a self-equilibrating stress produced by a nonuniform distribution of temperature or by differing thermal coefficients of expansion. Thermal stress is developed in a solid body whenever a volume of material is prevented from assuming the size and shape that it normally would under a change in temperature. Evaluation of thermal stress is not required by this Subsection."

This guidance is repeated in note 5 below Table NF-3523(b)-1, which states:

> "Thermal stresses within the support as defined by NF-3121.11 need not be evaluated."

Thus, in accordance with ASME Code, Section III, Subsection NF, requirements for Class 3 component supports, the thermal stresses in the SFP rack are not combined with the primary stresses due to dead (D), live (L), and seismic loads (E, E'). However, the allowable stress limits for each load combination (for example, D + L + To + E) are computed based on the yield and ultimate tensile strengths of the material at a temperature that bounds the temperature distribution in the SFP rack under To and Ta, as applicable.

In addition to the load combinations defined in Table 1 of Appendix D to SRP Section 3.8.4, SRP Section 3.8.4 also states:

> "The temperature gradient across the rack structure that results from the differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure."

The above requirement is the genesis for Section 6.8.2 of Attachment 5 & 6 to the LAR. The purpose of Section 6.8.2 is to (a) quantify the thermal gradient across the boundary between a full and empty cell and (b) ensure that cell-to-cell welds are strong enough to withstand the shear stresses induced by the temperature gradient.

In summary, the structural design and qualification, of the replacement SFP racks at PNP, follows the latest NRC staff guidance on spent fuel racks as stated in SRP Section 3.8.4.

NRC Request

5. *Section 6.7.4 of Attachment 6 to the LAR submittal presents a summary of the rack-to-rack and rack-to-wall impacts. The discussion in this portion of the submittal focuses on the rack-to-rack impacts at the base of the racks, however no discussion is provided regarding potential impacts at the top of the racks. Please discuss whether the time-history analyses performed using DYNARACK resulted in any apparent rack-to-rack impacts at the top of the rack structures. Additionally, if these impacts were predicted, confirm that the consequences of these impacts (e.g., buckling) have been sufficiently accounted for in the overall analysis of the racks.*

ENO Response

5. The time history analyses performed using DYNARACK did not result in any rack-to-rack impacts at the top of the rack structures. This is explained by the fact that the maximum predicted displacement at the top of the replacement SFP racks is only 0.3028-inch (refer to Table 6.6.2 of Attachment 5 & 6 to the LAR), whereas the minimum clearance gap to either a wall or an adjacent SFP rack is 0.625-inch. For the new SFP rack in the north tilt pit, the maximum predicted displacement at the top of the rack is 0.6952-inch (refer to Table 6.6.2 of Attachment 5 & 6 to the LAR), as compared to a minimum clearance gap of 5.04-inch.

NRC Request

6. *Figure 6.4.1 of Attachment 6 to the LAR submittal depicts the model used for DYNARACK time history analyses. Please state whether the DYNARACK model has been previously benchmarked against other numerical analysis methods, namely methods which employ a more explicit method of analysis (e.g., commercial finite element analysis software). Additionally, state whether the Palisades-specific acceleration time histories were used to benchmark the DYNARACK time history analyses.*

ENO Response

6. The DYNARACK model has been previously benchmarked against a three-dimensional finite element model created in LS-DYNA, a commercial finite element analysis code. This benchmark work is described in a previous RAI response related to the LAR submittal for the Beaver Valley Unit 2 rack replacement (FirstEnergy letter to the NRC dated May 21, 2010, Docket No. 50-412, number L-10-151, ADAMS Accession number ML101460057), which is reproduced below in its entirety. PNP-specific acceleration time histories were not used to benchmark the DYNARACK time history analyses. The benchmark work described below adequately demonstrates the capability of the DYNARACK model to conservatively predict the seismic response of an HCC SFP rack.

The 3-D lumped mass single rack model is the basic building block for whole pool multi-rack (WPMR) analysis, which is carried out using the Holtec proprietary code DYNARACK. The DYNARACK code was developed in the late 1970s and has been periodically updated since that time to incorporate technology advances such as multi-body fluid coupling which is a computer code based on the Component Element Method (CEM) (Reference 5). The chief merit of the CEM is its ability to simulate friction, impact, and other nonlinear dynamic events with accuracy. The high density racks designed by Holtec International are ideally tailored for the CEM-based code because of their honeycomb construction (HCC). Through the interconnection of the boxes, the HCC rack essentially simulates a multi-flange beam. The beam characteristics of the rack (including shear, flexure, and torsion effects) are appropriately modeled in DYNARACK using the classical CEM "beam spring". Each rack is modeled as a prismatic 3-D structure with support pedestal locations and the fuel assembly aggregate locations set to coincide with their respective center of gravity (C.G.) axes. The rattling between the fuel and storage cells is simulated in exactly the same manner as it would be experienced in nature: namely, impact at any of the four facing walls followed by rebound and impact at the opposite wall. Similarly, the rack pedestals can lift off or slide as the instantaneous dynamic equilibrium would dictate throughout the seismic event. The rack structure can undergo overturning, bending, twist, and other dynamic motion modes as determined by the interaction between the seismic (inertia) impact, friction, and fluid coupling forces.

Figure 5.1 and figure 5.4 of Holtec report HI-2084175 (Enclosure C to Reference 1) depicts the flexible elements used to model the dynamic behavior of the rack modules. The elements allow shear and bending deformation in each of the two horizontal directions perpendicular to the face of the racks. Additional elements are included in the model to allow axial deformation and torsional rotation.

The stiffness values were determined by considering each rack module as a beam with multiple flanges and webs comprised of the rack cell walls. The stiffness values were computed by deriving the appropriate formula from the Principle of Complementary Energy found in Advanced Mechanics of Materials (Reference 11). The resulting stiffness values accurately reproduce the stiffness matrix for a beam.

The computed rack stiffness values indicate considerable rigidity within the rack module. Consequently, the rack exhibits primarily rigid body motion during the dynamic earthquake event. The selection of six degrees of freedom (three translations and three rotations) at the top of the rack and of six degrees of freedom at the bottom of the rack provides adequate representation of the rigid body motion and captures first mode elastic response.

Ten additional degrees of freedom are added to represent fuel rattling within the storage rack cells. The fuel assembly mass represents the largest component of the dynamic rack-fuel system model and dominates the behavior of a filled storage rack module. Therefore, the entire assemblage is comprised of 22 degrees of freedom (dof), which are adequate to represent the dynamic behavior of each rack module.

This modeling construct has been used consistently by Holtec for more than two decades to analyze spent fuel racks on numerous dockets, and it has been reviewed extensively by the Atomic Safety and Licensing Board (ASLB), the NRC staff, and NRC consultants (Brookhaven National Laboratory and Franklin Research Center). Notwithstanding past history, a benchmark comparison between the 22-dof single rack model and a detailed finite element model has been performed to further demonstrate the adequacy of the simplified mass model to predict the anticipated time history seismic responses.

In 2008, Holtec developed a detailed finite element model of a 12 x 12 PWR spent fuel rack (Reference 6) for Sizewell nuclear plant. This nuclear power plant is located in England and operated by British Energy. The model, which is shown below in the Figure RAI-20-1, was built using LS-DYNA.

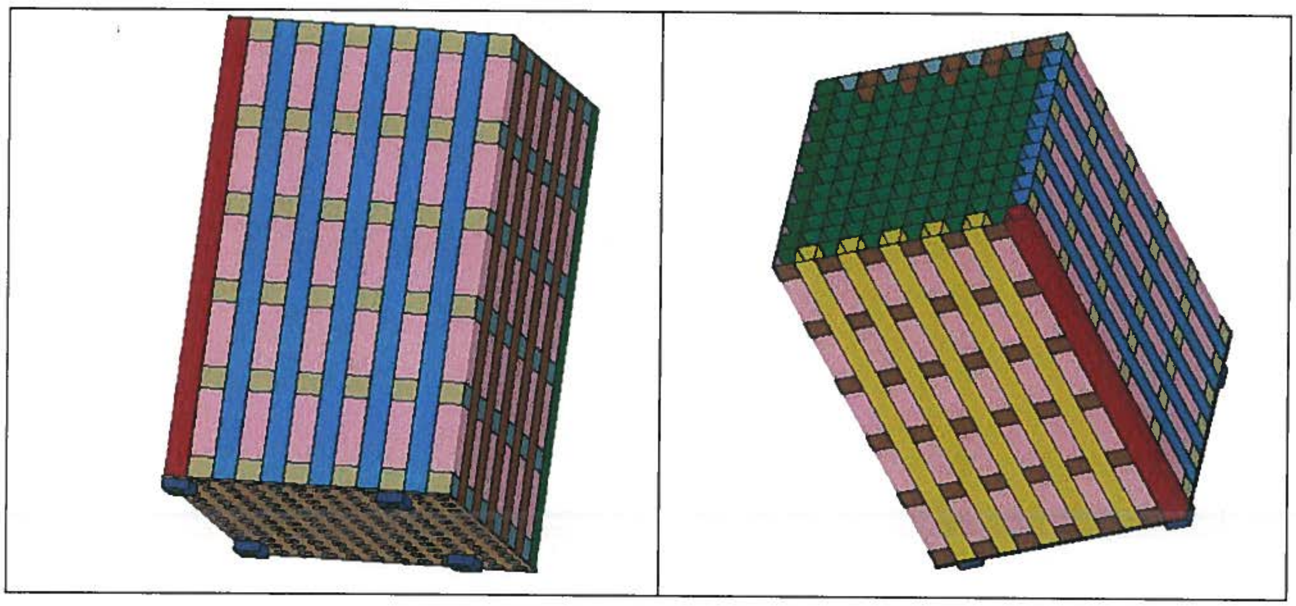

Figure RAI- 20-1 LS-DYNA Model of 12 x 12 Sizewell Rack

As shown in the figure above, all load bearing components of the spent fuel rack, including the support pedestals, the base plate, the cell walls, and the weld connections, were modeled explicitly using a combination of thick shell, shell, and solid elements. In addition to the rack structural components, the stored fuel assemblies were individually modeled in LS-DYNA as separate bodies and positioned within every storage cell location. Appropriate contact interfaces were defined between the cell walls, the base plate, and the fuel assemblies to allow each fuel assembly to rattle freely inside its storage cell under the applied loading. The entire model included nearly 200,000 elements.

A seismic analysis of the 12 x 12 Sizewell rack was performed by simultaneously applying three orthogonal acceleration time histories to the LS-DYNA model. For this analysis, the spent fuel rack was assumed to be freestanding in air (rather than in water) since it is difficult to model fluid-structure interactions in LS-DYNA directly. To offset this conservative assumption, an attenuation factor was applied to the horizontal input accelerations to account for the effect of the water.

In parallel with the LS-DYNA work, a separate analysis was also performed using DYNARACK for the same 12 x 12 Sizewell rack in air, under the same earthquake and fuel assembly loading conditions. It is further noted that the Sizewell rack was modeled in DYNARACK using the standard modeling approach (i.e., two-node, single beam rack model). Table RAI-20-1 summarizes the results predicted by both models.

Table RAI-20-1 Benchmarking for 12 x 12 Sizewell Rack

Result	LS-DYNA Model	DYNARACK Model
Max. Horizontal Displacement at Top of Rack (inches): North-South Direction East-West Direction	 0.761 0.999	 1.526 1.772
Max. Horizontal Displacement at Base of Rack (inches): North-South Direction East-West Direction	 0.216 0.228	 1.132 0.929
Maximum Instantaneous Fuel-to-Cell Impact Load (per assembly average) (lbf):	624	> 859

As expected, the DYNARACK model predicts larger rack displacements than LS-DYNA mainly because of the conservative manner in which rattling fuel assemblies are modeled in DYNARACK. The DYNARACK model assumes that all fuel assemblies rattle in-phase by virtue of the fact that the total mass of the stored fuel assemblies is lumped into a single fuel column. By contrast, the fuel assemblies are individually modeled in LS-DYNA, and therefore they are capable of rattling out-of-phase (as would actually occur in a seismic event). Out-of-phase rattling reduces the net impact between the stored fuel and the spent fuel rack, which in turn leads to smaller rack displacements. Thus, the simplified mass model utilized in DYNARACK provides a conservative prediction of the maximum rack displacements and fuel-to-cell impact load.

Even though the benchmarking was performed for a Sizewell rack, the conclusion is valid for nearly all Holtec designed spent fuel racks (including BVPS-2) since there is minimal change in the fabrication materials and the cell geometry across rack designs for different plants. To emphasize this point, the key characteristics of the 12 x 12 Sizewell rack, which is the focus of the benchmarking study, and the BVPS-2 spent fuel racks are summarized in the following table.

Table RAI-20-2

	Sizewell	BVPS-2
Cell wall material	SA-240 304	SA-240 304/304L
Cell height above base plate (in)	168.625	169
Cell wall thickness (in)	0.075	0.075
Cell inside dimension (in)	8.87	8.80
Cell pitch (in)	9.07	9.03

Based on the data in Table RAI-20-2 and the size of the rack (e.g., 12 x 12 or 10 x 14), the equivalent beam stiffness values for the rack cell structure are computed for input to the two-node, single beam DYNARACK model. The following table summarizes the stiffness values for the 12 x 12 Sizewell rack and the 10 x 14 BVPS-2 rack as analyzed in Reference 3.

Table RAI-20-3

	12 x 12 Sizewell Rack	10 x 14 BVPS-2 Rack
Shear Stiffness in X Dir. (lbf/in)	3.200E+06	2.788E+06
Shear Stiffness in Y Dir. (lbf/in)	3.200E+06	3.327E+06
Axial Stiffness (lbf/in)	3.410E+07	3.305E+07
Bending Stiffness about X Axis (lbf-in/rad)	3.690E+10	4.750E+10
Bending Stiffness about Y Axis (lbf-in/rad)	3.690E+10	2.500E+10
Torsional Stiffness (lbf-in/rad)	3.973E+09	3.671E+09

To further establish the link between the Sizewell benchmark study and the spent fuel racks to be installed at BVPS-2, four additional DYNARACK runs have been performed. The following table summarizes the key parameters for each run.

Table RAI-20-4

Run No.	Rack Analyzed	Surrounding Fluid	Input Earthquake	Fuel Assemblies
1	12 x 12 Sizewell Rack	Water	Sizewell SSE	Sizewell
2	10 x 14 BVPS-2 Rack	Water	Sizewell SSE	Sizewell
3	12 x 12 Sizewell Rack	Water	BVPS-2 SSE	BVPS-2
4	10 x 14 BVPS-2 Rack	Water	BVPS-2 SSE	BVPS-2

Run Numbers 1 and 2 are developed to show that both the Sizewell rack and the BVPS-2 rack respond similarly when submerged in water and subjected to the Sizewell design basis earthquake. Run Numbers 3 and 4 are developed to show that both the Sizewell rack and the BVPS-2 rack respond similarly when submerged in water and subjected to the BVPS-2 design basis safe shutdown earthquake (SSE). The results from these four runs are summarized below in Table RAI-20-5.

Table RAI-20-5

Result	Run No. 1	Run No. 2	Run No. 3	Run No. 4
Max. Horizontal Displacement at Top of Rack (in):				
X Direction	0.889	0.799	0.803	0.702
Y Direction	0.861	0.465	0.767	0.394
Max. Horizontal Displacement at Base of Rack (in):				
X Direction	0.023	0.041	0.112	0.069
Y Direction	0.028	0.023	0.182	0.048
Maximum Fuel-to-Cell Impact Load (per assembly average) (lbf):	813	586	729	717

To summarize, the two-node, single beam rack model implemented in DYNARACK has been compared against a detailed LS-DYNA finite element model, and as shown in Table RAI-20-1, DYNARACK provides a more conservative prediction of the time history seismic response. Although the benchmark comparison is performed for a 12 x 12

Sizewell rack, the results are valid for nearly all Holtec designed spent fuel racks, including those to be installed at BVPS-2, because of the similarities in fabrication materials and cell geometry. The similarity between the 12 x 12 Sizewell rack and the BVPS-2 spent fuel racks is further demonstrated by the DYNARACK results in Table RAI-20-5.

References

1. Letter from P. P. Sena, FirstEnergy Nuclear Operating Company, to NRC Document Control Desk, "Beaver Valley Power Station, Unit No. 2 Docket No. 50-412, License No. NPF-73, License Amendment Request No. 08-027, Unit 2 Spent Fuel Pool Rerack," with Enclosure B (proprietary) and Enclosure C (non-proprietary), "Licensing Report for Beaver Valley Unit 2 Rerack," dated April 9, 2009. (ADAMS Accession Nos. ML091210251 (letter) and ML091210263 (Enclosure C))

3. Holtec Report HI-2084165, "Structural Evaluation of Cask Pit Platform", Revision 4.

5. Levy, S. and Wilkinson, J., The Component Element Method in Dynamics, McGraw-Hill, Inc., 1976.

6. Holtec Report HI-2084009, "Seismic/Structural Analysis of Spent Fuel Rack D4 at Sizewell Nuclear Plant", Revision 1.

11. Boresi, P. et al., Advanced Mechanics of Materials, John Wiley & Sons, Inc., Fifth Edition.

NRC Request

7. *With respect to the numerical analyses performed for the fuel handling accidents, as described in Section 7 of Attachment 6 to the February 28, 2012, LAR submittal, please address the following as they relate to the use of the LS-DYNA computer code to perform these analyses:*

   a. *Table 7.4.2 of Attachment 6 of the LAR submittal for the Palisades re-rack documents the data points used for the strain rate amplification curve applicable to the base metal material for the LS-DYNA fuel handling accident simulations. Provide the bases for the strain rate amplification data and any references which document prior NRC approval regarding the use of this data.*

   b. *Table 7.3.2 of Attachment 6 documents the material properties used in the numerical analyses for the racks. Please provide these properties for the rack weld material as they are applied in the LS-DYNA simulation and state the bases for these properties.*

ENO Response

7.a The strain rate data is obtained from the test results reported in the literature--- D. K. Morton et. al., "Impact Testing of Stainless Steel Material at Room and Elevated Temperatures," 2007 ASME Pressure Vessels and Piping Division Conference. The same strain rate data was used in the Beaver Valley Unit 2 SFP replacement rack LAR (see "Supplement to RAI 17" in Reference 1), which was approved by the NRC.

<u>References</u>

1. FirstEnergy Nuclear Operating Company letter L-10-001 to the NRC, "Response to Request for Additional Information for License Amendment Request No. 08-027, Unit 2 Spent Fuel Pool Rerack (TAC No. ME1079)" dated January 18, 2010.

7.b The weld material is conservatively assumed to have the same material properties as the base metal (i.e., SA240-304L) in the LS-DYNA simulation. Thus, the properties for the rack weld material are the same as those listed in Table 7.3.2 for "Austenitic Stainless Steel." The property values listed in Table 7.3.2 are the minimum required strength properties of SA-240 304L as specified in the ASME Code, Section II, Parts A and D. It is noted that the material properties in Table 7.3.2 are engineering stresses and strains (as opposed to true stresses and strains). The use of engineering stress/strain values as input to LS-DYNA is conservative because it underestimates the strain energy capacity of the weld material at its failure limit.

NRC Request

8. Section 7.2 of Attachment 6 of the LAR submittal for the Palisades re-rack summarizes the fuel handling accidents considered in the structural design of the Palisades replacement SFP storage racks. The shallow drop accident considers the case whereby a fuel bundle and handling tool are postulated to drop on an outer rack cell wall. For the shallow drop accident, please provide a technical justification which confirms that the outer cell wall produces the limiting drop location, as opposed to a drop of the bundle over an interior portion of the SFP rack.

ENO Response

8. The replacement SFP racks at PNP are Region 1 racks, as described in Section 2.1 of Attachment 5 & 6 to the LAR. Each storage cell location in the SFP rack is formed from a composite box assembly (see Figures 2.6.1 and 2.6.3 of Attachment 5 & 6 to the LAR). Consequently, the outer rack cell wall is more limiting because (a) only a single cell wall is available to resist the shallow drop

impact and (b) there are fewer cell-to-cell connections along the perimeter of the rack as compared to the interior portion of the SFP rack.

Accident Dose Branch
NRC Request

1. On page 13 of Attachment 1 to your February 28, 2012, submittal, it states:

The structural damage to the fuel building, pool liner, and fuel assembly resulting from a dropped fuel assembly striking the pool floor or another assembly located in the racks is primarily dependent on the mass of the falling object and drop height. Since these two parameters are not changed by the proposed modification, the postulated structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary has been evaluated and found to remain well below levels established by regulatory guidance.

Please provide additional information for the basis that the radiological dose at the exclusion area boundary was found to remain well below the levels described in regulatory guidance. Include the following information in your response:

a. Provide which regulatory guidance that is being referenced.

b. Provide assumptions and parameters used in the above mentioned radiological dose analyses.

c. Provide the depth (i.e. how many feet) of water coverage is assumed in the above mentioned radiological dose analyses.

d. Provide the decontamination factor assumed in the above mentioned radiological dose analyses.

e. Provide the resulting radiological dose values that your conclusion is based upon including the low population zone and control room values.

ENO Response

1. Structural damage to the fuel building, pool liner, and fuel assemblies, resulting from a dropped fuel assembly striking the pool floor or another assembly located in the racks, is considered bounded by the design basis evaluation of a postulated cask drop accident.

As described in the PNP FSAR, Section 14.11, "Postulated Cask Drop Accidents," in 2003, a PNP facility change modified the main hoist of the fuel building crane to increase the capacity to 110-tons, and to meet single failure criteria in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear

Power Plants," and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Since the main hoist has been upgraded to meet single-failure-proof criteria, analyses of postulated load drops from the main hoist are no longer required.

However, to address potential situations in which lifting devices used with the main hook do not meet single failure requirements, single-failure-proof features of the main hoist are disabled, or lifts with the 15-ton auxiliary hoist fail to follow designated safe load paths, several structural and radiological analyses are retained and referenced in the FSAR, Section 14.11, for cask drops in the spent fuel pool area.

Radiological consequences resulting from a dropped fuel assembly striking the pool floor or another assembly located in the racks is considered bounded by the design basis evaluation of the postulated cask drop accident and the fuel handling accident.

A summary of the PNP cask drop accident radiological design basis analysis is contained in the NRC safety evaluation for the alternative source term. Letter from NRC to ENO, entitled "Palisades Nuclear Plant – Correction Letter for Alternative Radiological Source Term Amendment," dated May 7, 2008 (TAC No. MD3087). Refer to section 3.1.7, page 31, of the safety evaluation. (ADAMS accession number: ML080660133).

In summary:

a. The analysis conforms to Regulatory Guide 1.183.

b. Major assumptions include:
   - Complete damage to 73 fuel assemblies stored in the spent fuel pool.
   - Gap release fractions for I-131 and Kr-85 adjusted as specified by Table 3 of RG 1.183.
   - Per RG 1.183, Appendix B, Regulatory Position 1.3, iodine release is comprised of 99.85% elemental iodine and 0.15% organic iodine.
   - Three cases evaluated:
     - Case 1 with 90% of the release via the FHB filtration system, 30 days of decay, and the control room initially aligned in emergency recirculation mode.
     - Case 2 with 82.5% of the release via the FHB filtration system, 30 days of decay, and the control room initially aligned in emergency recirculation mode.
     - Case 3 with 0% of the release via the FHB filtration system, 90 days of decay, and no isolation of the control room.
   - Per RG 1.183, Appendix B, Regulatory Position 4.1, the release occurs over a two-hour period.

c. A minimum of 23.4 feet of water is maintained above fuel stored in the spent fuel pool fuel.

d. Per RG 1.183, Appendix B, Regulatory Position 2, an overall iodine decontamination factor of 200 is used.

e. Resulting radiological doses below are taken from FSAR Table 14.1-6:

FSAR SECTION	SCENARIO DESCRIPTION			OFFSITE DOSES AND LIMITS				CONTROL ROOM HABITABILITY			
	Cask Drop Scenario	Pre-Isolation % Filter Bypass	Post-Isolation % Filter Bypass	Offsite Dose Limits (5) [rem]		Exclusion Area Boundary (0-2 hrs) [rem]	Low Population Zone [rem]	Time to E-HVAC [min] / CRE Inleakage [cfm]	Control Room Dose Limits (6) [rem]		Control Room Dose [rem]
Section 14.11: CASK DROP IN THE SPENT FUEL POOL (Section 14.24 for CRH)	30 Day Decay w/Charcoal Filters	N/A	10	TEDE	6.3	2.04	0.25	0 min / 100 cfm	TEDE	5	1.37
	30 Day Decay w/Charcoal Filters	N/A	17.5	TEDE	6.3	2.78	0.35	0 min / 100 cfm	TEDE	5	1.99
	90 Day Decay No Charcoal Filters	100	100	TEDE	6.3	0.08	0.01	Not Required (3) / 100 cfm	TEDE	5	1.67

NRC Request

2. Please provide a comparison of the above mentioned radiological dose analyses to the design-basis FHA dose analysis for Palisades Nuclear Plant.

ENO Response

2. For comparison, a summary of the PNP fuel handling accident radiological design basis analysis (described in PNP FSAR, Section 14.19) is contained in the NRC safety evaluation correction letter for the alternative source term. Letter from NRC to ENO, entitled "Palisades Nuclear Plant – Correction Letter for Alternative Radiological Source Term Amendment (TAC No. MD3087)," dated May 7, 2008. Refer to section 3.1.2, page 17, of the safety evaluation. (ADAMS accession number: ML080660133)

In summary:

a. The analysis conforms to Regulatory Guide 1.183.

b. Major assumptions include:
- Complete damage to one fuel assembly (216 fuel pins).
- Two days of decay following discharge from reactor core.

- Per RG 1.183, Appendix B, Regulatory Position 1.3, iodine release is comprised of 99.85% elemental iodine and 0.15% organic iodine.
- Three cases reported:
  - Base Case with 100% of the release via the plant equipment hatch
  - Case with 10% of the release via the FHB filtration system
  - Case with 50% of the release via the FHB filtration system
- Per RG 1.183, Appendix B, Regulatory Position 4.1, the release occurs over a two-hour period.

c. A minimum of 22.5 feet of water is maintained above damage fuel, since the fuel handling accident could occur in the refueling cavity in addition to the spent fuel pool.

d. An overall iodine decontamination factor of 183 is used. Based on the method of Burley (G. Burley, "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident", October 5 1971), with a pool depth of 22.5 feet, the decontamination factor for organic iodine is one and the decontamination factor for elemental iodine is 252, resulting in an overall iodine decontamination factor of 183.

f. Resulting radiological doses below are taken from FSAR Table 14.1-6:

FSAR SECTION	SCENARIO DESCRIPTION	OFFSITE DOSES AND LIMITS				CONTROL ROOM HABITABILITY			
	Dose Calculation Assumptions	Offsite Dose Limits (5) [rem]		Exclusion Area Boundary (0-2 hrs) [rem]	Low Population Zone [rem]	Time to E-HVAC [min] / CRE Inleakage [cfm]	Control Room Dose Limits (6) [rem]		Control Room Dose [rem]
Section 14.19: FUEL HANDLING INCIDENT (Section 14.24 for CRH)	One fuel bundle fails 2 days after shutdown, No Charcoal Filtration	TEDE	6.3	2.20	0.28	20 min / 100 cfm	TEDE	5	4.04
	One fuel bundle fails 2 days after shutdown, 10% Filtered Release	TEDE	6.3	2.02	0.25	20 min / 100 cfm	TEDE	5	3.68
	One fuel bundle fails 2 days after shutdown, 50% Filtered Release	TEDE	6.3	1.31	0.17	20 min / 100 cfm	TEDE	5	2.22

<u>Human Performance</u>

NRC Request

1. *In section 4.0 of Attachment 1 (Technical Analysis) in the Human Performance section, the LAR states that the human performance tools used during fuel handling activities are "considered appropriate to minimize the probability of the occurrence of a fuel misload event. Following the replacement of all Region I fuel storage racks, a misloading event within Region I could not occur." There is not*

enough information in the LAR to support the conclusion that a misload event could not occur once the fuel racks have been replaced. Please provide additional information which Palisades used to determine that a misloading event could not occur.

ENO Response

1. A misload event is one where a fuel assembly beyond the capability of the racks is loaded into a storage cell. As stated in Subsection 4.1 of Holtec Report HI-2115004, the new Region 1 racks are designed and qualified to store fresh (unburned) fuel assemblies with an initial enrichment of up to 5.0 weight percent. This fuel has a reactivity that bounds any fuel assembly previously, currently or intended to be used in the future at PNP. Such fuel may be placed into 100% of the storage locations in the new Region 1 racks. As all possible storage locations in the new racks are qualified for storage of any possible PNP fuel assembly, it is not possible to have a misload event.

   It is noted that the existing Region 1 racks rely on administrative controls to maintain an acceptable reactivity through the use of empty rack storage locations. Thus, the removal of the existing Region 1 racks and installation of the new Region 1 racks will eliminate the current potential for a misload event.

NRC Request

2. *The LAR does not indicate whether the proposed Technical Specification changes will present any new or increased opportunities for operator error. Please provide information as to whether the new changes might present increased opportunities for operator error, and if there are administrative controls in place to prevent or mitigate such errors.*

ENO Response

2. The design of the new Region 1 racks was performed with one of the primary goals being to avoid any new or increased opportunities for operator error. To this end, the following actions were taken during the design of the new racks:

   - The seating surfaces for the fuel assemblies in the new Region 1 racks were made essentially the same as those in the existing Region 1 racks, both in terms of elevation above the spent fuel pool floor and with respect to the holes that interface with the pins that project from the bottom of the fuel assemblies.

   - The overall length of the new Region 1 racks storage cells above the fuel seating surface was made essentially the same as those in the existing Region 1 racks, so that the elevation of cell top edges is essentially unchanged.

- The top ends of the new Region 1 racks storage cells were formed into a flared lead-in, similar to the existing Region 1 racks, to provide a similar level of assistance with fuel assembly insertion.

- The new Region 1 racks are approximately the same overall size as the existing Region 1 racks, so that no significant change in the rack-to-wall or rack-to-rack gaps are created.

The resulting new Region 1 racks should be able to be loaded and operated without needing any modifications to fuel handling equipment design and with minimal changes to fuel handling procedures. In addition, elimination of the administrative controls that apply to fuel loading for the existing Region 1 racks actually reduces the potential for operator error.

NRC Request

3. Section 4.0 of Attachment 1 contains very little information regarding Human Performance review activities. This RAI contains several questions:

1. Was an operating experience review done to consider lessons learned? And if so, please provide this information.
2. Have there been changes to spent fuel loading training? Please provide any information regarding changes to training or qualifications as a result of this LAR.
3. Please describe any changes to physical interfaces, such as monitoring instruments for radioactivity, boron concentration, and include any changes to the crane operation.
4. Will there be any changes required to the procedures for fuel movement, including engineering procedures for analyzing and planning moves? If so, please provide a list of those changes.

ENO Response

3.1. There was no specific human performance review of operating experience done that considered lessons learned at other plants.

The operating experience at PNP includes recent past (Renewed Facility Operating License (RFOL) Amendment No. 236 issued on February 6, 2008) and current loading pattern restrictions (RFOL Amendment No. 246 issued on January 27, 2012) in Region I of the SFP and loading restrictions in Region II (License Amendment Nos. 105 issued on July 24, 1987, 189 conversion to ITS on November 30, 1999, and 207 issued on February 26, 2002). The loading restrictions in Region II would remain in place with approval of the LAR. The loading pattern restrictions in Region I would remain in place while the current Region I Carborundum equipped storage racks remain in the SFP. The loading pattern restrictions in Region I would be removed by the proposed LAR, after all the Carborundum equipped storage racks are removed. The proposed change states, as noted above, that the new Metamic SFP storage racks are designed to accommodate unrestricted storage of fresh or spent fuel with an initial maximum nominal planar average U-235 enrichment of 5.0 wt%. Also stated in the LAR is that the current TS requirements, for the Region I Carborundum equipped fuel storage racks, limit fuel assemblies to a maximum planar average U-235 of 4.54 wt%, and restrict the fuel loading in designated sub-regions within Region I. Restrictions in the fuel loading in Region I began after testing identified degradation in the Carborundum neutron absorber as described in the Background section 3.0 of Attachment 1 in the LAR. Replacement of the storage racks would eliminate the fuel loading restrictions in Region I, and reduce the potential for errors by eliminating the potential for misloading of fuel in the new racks.

3.2. There will be no specific changes to SFP loading training or qualifications as a result of this LAR. During the installation of the new Metamic equipped fuel storage racks, current restrictions will remain for loading of fuel in the Carborundum equipped fuel storage racks. Current training and qualifications appropriately address the current loading restrictions. The unrestrictive loading requirements for the new Metamic equipped fuel storage racks would not require any specific training or qualifications changes.

3.3. There would not be any changes to physical interfaces, including monitoring instruments, boron concentration, or crane operation as a result of this LAR.

3.4. Changes to the fuel movement procedures as a result of this LAR would include:

- Steps to ensure two empty storage rows are maintained between the fuel assemblies in the Carborundum equipped storage rack and an adjacent Metamic equipped rack.

- Upon completion of the removal of the Region I Carborundum equipped storage racks from the SFP, procedure instructions associated with the current loading pattern restrictions of the Carborundum equipped racks will be removed.

Health Physics
NRC Request

1. What is the total person-rem estimated for the job. Include in the estimate, the dose to install the racks and the dose from the radioactive waste processing of the existing contaminated racks. If you have contingency plans to use divers, include the dose to the divers in the total estimate.

ENO Response

1. The total rem estimate for the job is 3.85 Rem.

PALISADES DOSE ESTIMATE					
Task	Quantity	No. of workers	Dose Field (mr/hr)	Hours	Total Dose (mr)
Rack Removal	7	6	5	10	2100
Fuel Assembly Liberation	11	4	2	10	880
Rack Installation	7	6	2	6	504
Fuel Shuffling	2	2	2	45	360
				Total	3.85 Rem

NRC Request

2. *Provide specific details of radiological controls for this job. Discuss how work, personnel, traffic and equipment movement are going to be monitored and controlled to minimize contamination and maintain exposure ALARA, protective clothing requirements, personnel monitoring requirements. What provisions will be implemented to detect, control and minimize worker exposure to discreet radioactive particles in the work environment that may be generated or displaced by fuel rack replacement activities?*

ENO Response

2. By gaining an understanding of the project in its entirety and focusing on specific dose reduction and contamination control issues, each worker can effectively lower their accumulated dose. Nominal man-hours along with As Low As Reasonably Achievable (ALARA) total effective general area dose rates are the key determining factors to ensure this project meets the estimated dose assigned by the ALARA committee. This will be accomplished by using installed temporary shielding, keeping water activity levels as low as possible and wrench time man-hours to a minimum. General area dose rates rise as the radioactive material within the SFP is brought to the water surface. It is critical that the project team capture the insoluble water activity as it gets agitated due to pressure washing and moving items around in the SFP. Minimal general area dose rates will also be achieved by thoroughly flushing/pressure washing the old SFP racks along with other items removed from the pool water. There is a direct correlation between the more activity fixed on the rack/component vs. the associated general area dose rate at the rack/component due to the source size. Therefore, if the rack activity and associated dose rate is high, more dose will be accumulated by the work crew while bagging/shipping the rack vs. low general area dose rate/activity.

Installation of new racks involves the following activities: Establish and maintain a Foreign Material Exclusion (FME) zone and monitor in a low dose rate area/ALARA zone away from the Highly Contaminated Area (HCA) in an effort to maintain all workers dose ALARA. The Radiation Protection (RP) staff will position Wireless Remote Monitors (WRM), and/or other area radiation monitors along with air monitors to verify general area dose rates and verify air quality /contamination control. Additional remote monitoring may be positioned to verify conditions on high dose rate components, filters and the underwater debris basket. RP will set up the area with designated contaminated areas/HCAs, as needed, dependent on the Holtec laydown/storage areas.

General area alarming air samplers will be in place during the project. Alarming radiation monitors will be positioned in strategic areas of the work site. Additional low volume air sampling units (RAS pumps) may also be utilized.

Throughout the work evolution, water activity and the spent fuel pool boron concentration levels will be monitored by chemistry staff. A base line dose rate trending survey will be performed: Pre and post-work area β/γ dose rate surveys, contamination $\beta/\gamma/\alpha$ surveys, along with air sample surveys will be performed during the project in addition to the continuous coverage support. This data will be correlated with the SFP water activity analysis performed by chemistry staff and tracked by RP to verify that worker general area dose rates remain ALARA.

Keeping associated person-hours assigned to the project ALARA can be accomplished through minimal re-work and/or un-necessary handling of radioactive components. Each worker will be expected to utilize good radworker practices, three way communication, and a questioning attitude to maintain overall project dose ALARA.

NRC Request

3. *Are divers being considered either to perform any activity or being considered as a contingency plan? If divers are being considered, provide specific detail of the radiological controls consistent with the requirements of Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas of Nuclear Plants."*

ENO Response

3. Divers are not being considered for performing any activity.

NRC Request

4. *Provide the necessary information (e.g., shielding calculations, plant layout, etc.) that demonstrates that the radiological impact to areas of the plant adjacent to the spent fuel pool from the proposed design change is minimal. Is the change expected to impact the plant radiation zoning design?*

ENO Response

4. There is no expected change in the plant radiation zoning design. The type of fuel in the SFP is not changing and the elevation of the fuel in the SFP racks is not changing.

NRC Request

5. *Discuss the methods that will be used to remove radioactive crud, sediment and other debris generated in the rack replacement and maintain water clarity in the pool. Is the additional quantity of solid radioactive waste generated from the requested changes expected to result in a significant change in the generation of solid radioactive waste at PNP?*

ENO Response

5. During the course of rack installation and removal work at PNP, there exists the possibility of generation or dislodging of some radioactive and non-radioactive crud, sediment, and/or debris. This material typically accumulates on the existing rack module surfaces, fuel assemblies, or beneath the existing racks and may be disturbed by extensive fuel shuffles and rack movements. Several actions are taken to protect against increases in personnel dose rates and operability reductions from pool clarity. Prior to removal of racks, each of the existing cell locations will be pressure washed on the four internal faces to dislodge potential radioactive solid waste that may have adhered to the internal surfaces. At times, multiple passes, as determined by dose surveys, may be required to reduce dose rates to an acceptable level for removal.

   In addition to dislodged waste or materials during preparation and handling of SFP racks, the existing racks themselves will become radioactive waste. In total, six large racks weight approximately 14 tons each and a single smaller rack weighing 11 tons will be removed from the SFP to be replaced by new racks. As a result, a total of approximately 95 tons of mostly metallic radioactive solid waste will be created (non-metallic portions comprised of rack materials such as Carborundum and miscellaneous pool settlement).

   During all operations except actual rack handling, existing plant equipment will be utilized to remove materials dispersed to the bulk SFP water. These typically include SFP cooling filtration, skimmers for surface materials, and supplemental "tri-nuke" pool vacuum equipment. At all times activity is monitored, so tracking of activity level trends as a result of pressure washing and rack manipulations is possible. During rack handling, SFP cooling must be off to reduce water movement to facilitate safe rack manipulation. In addition, there has been operational experience that degraded rack poison materials can decrease pool clarity and therefore inhibit rack handling. In such cases, one to two days of filtration by plant systems is typically able to restore sufficient clarity to continue rack manipulations for subsequent racks. After removal of existing racks, there will some settlement materials observed on the pool floor. To avoid additional disbursement of these materials, after each removal sequence (for example, removal of 3 racks) the floor will be vacuumed to remove bulk materials.

Steam Generator Tube Integrity and Chemical Engineering Branch
NRC Request

1. *Please provide the following:*

   a. *Physical dimensions (length, width, and thickness) of the coupons in the coupon surveillance program.*

   b. *A description of how the coupons mounted on the coupon tree are representative of the Metamic installed in the spent fuel pool cells. Will there be any sheathing covering the coupons similar to the sheathing holding the Metamic in the storage cells?*

   c. *Justification on how the coupon thermal and chemical environment will be similar to that of the Metamic in the storage cells.*

ENO Response

1.a. The coupons will be approximately 6" × 8" × 0.106".

1.b. The coupons are made by excising them from the same larger pieces of material as the Metamic panels installed in the racks. As such, they are chemically identical to the Metamic in the racks. The coupons are mounted on "trees" by way of threaded rods, which protrude from the trees and pass through holes in the coupons, washers and wing nuts. There is no sheathing over the coupons, to ensure the coupons are exposed to the SFP water and to minimize the shielding of gamma radiation that reaches the coupons.

1.c. The coupons are placed into storage cells of the SFP racks and are, therefore, exposed to the same SFP water temperature and chemical environment (including boron concentration) as is the Metamic installed in the fuel racks.

NRC Request

2. *The license amendment request (LAR) stated that over the duration of the coupon testing program, the coupons will have accumulated more radiation dose than the expected lifetime dose for the normal storage cells. This is based on the tree being placed in an area of the highest dose for the first four offloads. Please provide a description of how you will ensure that the coupons will have accumulated more dose than the Metamic in the storage cells for the life of the coupons following the first four offloads.*

ENO Response

2. The coupons will be installed in storage cells of the Region 1 racks, which will be where the fuel discharged from the reactor is placed during each outage. The coupon locations will specifically be surrounded by this freshly discharged fuel, which is the most emissive fuel in the SFP in terms of gamma radiation. Consistently surrounding the coupons with the most emissive fuel will maximize the local gamma fluxes in the vicinity of the coupons compared to the average gamma fluxes in the racks. Given the exponential nature of the decrease of gamma flux with increasing fuel cooling time, the maximum gamma flux the coupons will be exposed to will be much greater than the average flux that the Metamic in the other storage cells will be exposed to. As long as the coupons are exposed to a higher gamma flux at the beginning of the surveillance program, they will be leading indicators of the Metamic in the rest of the rack cells.

NRC Request

3. *The LAR stated coupons that were removed from the pool for testing that had not been destructively analyzed, may optionally be returned to the spent fuel pool and remounted.*

   a. *How long will a coupon be allowed to remain out of the pool before being reinserted?*
   b. *Will the coupon go through a vacuum drying phase before testing and reinsertion?*
   c. *Provide a justification for how the coupon will still be representative of the Metamic in the storage cells once it is reinserted.*

ENO Response

3.a. Plans are for intact coupons to be reinserted to the SFP within three months of their removal from the pool. Compared to the multi-year duration of the surveillance program, such a short duration outside of the pool should not be significant.

3.b. Metamic is a fully-dense material with no significant porosity, so water intrusion into the core of the panels is not possible. As the coupons will only have water on their surfaces, limited by the water surface tension, it is not necessary to resort to a vacuum drying process. Following removal of a coupon from the SFP, the coupon would be cleaned and then wiped dry to remove the surface water prior to testing.

3.c. As noted above in 3.a., plans are for intact coupons to be reinserted to the pool within three months of their removal. Compared to the multi-year duration of the surveillance program, such a short duration outside of the pool should not be significant. The reinserted coupons would not be used for normal testing under the surveillance program, but would act as contingency panels that could be used to confirm results from testing of coupons removed from the pool for the first time. This would permit any unexpected result from testing of a coupon to be confirmed without deviating from the normal testing schedule.

ATTACHMENT 2

Revised Technical Specifications Pages 4.0-4 and 4.0-5

(Replace pages 4.0-4 and 4.0-5 in the February 28, 2012, LAR)

Two pages follow

4.3 Fuel Storage

4.3.1 Criticality (continued)

3. Control blades may be stored in both fueled and unfueled locations in Regions 1D and 1E, with no limitation on the number.

4.3.1.2 The Region I (See Figure B 3.7.16-1) Metamic equipped fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. A nominal 10.25 inch center to center distance between fuel assemblies;

e. New or irradiated fuel assemblies;

f. Two empty rows of storage locations shall exist between the fuel assemblies in a Carborundum equipped rack and the fuel assemblies in an adjacent Metamic equipped rack; and

g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2.

4.3.1.3 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with:

a. Fuel assemblies having maximum nominal planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

4.3 Fuel Storage

4.3.1 Criticality (continued)

e. New or irradiated fuel assemblies which meet the maximum nominal planar average U-235 enrichment, burnup, and decay time requirements of Table 3.7.16-1.

4.3.1.4 The new fuel storage racks are designed and shall be maintained with:

a. Twenty four unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. Keff ≤ 0.95 when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.

ATTACHMENT 3

Mark-up of Technical Specifications Pages 4.0-4 and 4.0-5

(showing proposed changes; additions are highlighted and deletions are strikethrough)

Two pages follow

4.3.1 Criticality (continued)

3. Control blades may be stored in both fueled and unfueled locations in Regions 1D and 1E, with no limitation on the number.

4.3.1.2 The Region I (See Figure B 3.7.16-1) Metamic equipped fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowances for uncertainties as described in Section 9.11 of the FSAR;

d. A nominal 10.25 inch center to center distance between fuel assemblies;

e. New or irradiated fuel assemblies;

f. Two empty rows of storage locations shall exist between the fuel assemblies in a Carborundum equipped rack and the fuel assemblies in an adjacent Metamic equipped rack; and

g. A minimum Metamic B^{10} areal density of 0.02944 g/cm^2.

4.3.1.~~2~~3 The Region II fuel storage racks (See Figure B 3.7.16-1) are designed and shall be maintained with~~;~~:

a. Fuel assemblies having maximum nominal planar average U-235 enrichment of 4.60 weight percent;

b. Keff < 1.0 if fully flooded with unborated water, which includes allowances for uncertainties as described in Section 9.11 of the FSAR.

c. Keff ≤ 0.95 if fully flooded with water borated to 850 ppm, which includes allowance for uncertainties as described in Section 9.11 of the FSAR.

d. A nominal 9.17 inch center to center distance between fuel assemblies; and

4.3 Fuel Storage

4.3.1 Criticality (continued)

e. New or irradiated fuel assemblies which meet the maximum nominal planar average U-235 enrichment, burnup, and decay time requirements of Table 3.7.16-1.

4.3.1.~~3~~4 The new fuel storage racks are designed and shall be maintained with:

a. Twenty four unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.95 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1, or

Thirty six unirradiated fuel assemblies having a maximum nominal planar average U-235 enrichment of 4.05 weight percent, and stored in accordance with the pattern shown in Figure 4.3-1;

b. Keff ≤ 0.95 when flooded with either full density or low density (optimum moderation) water including allowances for uncertainties as described in Section 9.11 of the FSAR.

c. The pitch of the new fuel storage rack lattice being ≥ 9.375 inches and every other position in the lattice being permanently occupied by an 8" x 8" structural steel or core plugs, resulting in a nominal 13.26 inch center to center distance between fuel assemblies placed in alternating storage locations.

4.3.2 Drainage

The spent fuel storage pool cooling system suction and discharge piping is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 644 ft 5 inches.

4.3.3 Capacity

The spent fuel storage pool and north tilt pit are designed and shall be maintained with a storage capacity limited to no more than 892 fuel assemblies.

ATTACHMENT 4

HOLTEC INTERNATIONAL AFFIDAVIT

Five pages follow

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053
Telephone (856) 797-0900
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Holtec International Document ID 2119-AFFI-02R1

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Evrim K. Kalfazade, being duly sworn, depose and state as follows:

(1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld is information provided within Entergy Nuclear Operations, Inc. letter number PNP 2012-071, "Response to Request for Additional Information – License Amendment Request – Replacement of Spent Fuel Pool Region I Storage Racks," specifically in Attachment 5 that is the response to Reactor Systems (SRXB) item number 4, which contain Holtec Proprietary information and is appropriately marked as such.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used 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 of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraph 4.b, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary

agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Mr. Evrim K. Kalfazade, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 29th day of August, 2012.

Evrim Kalfazade

Evrim K. Kalfazade
Holtec International

Subscribed and sworn before me this 29th day of August, 2012.

Maria C Massi

MARIA C. MASSI
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2015