October 31, 2005

Mr. Christopher M. Crane, President
and Chief Executive Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT 1 - ISSUANCE OF AN AMENDMENT - RE: ONSITE SPENT FUEL STORAGE EXPANSION (TAC NO. MC4202)

Dear Mr. Crane:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 170 to Facility Operating License No. NPF-62 for the Clinton Power Station (CPS), Unit 1. The amendment is in response to your application dated August 18, 2004, as supplemented on May 13 and 25, June 14, August 17, and October 24 and 25, 2005.

The amendment revises Technical Specification 4.3, "Fuel Storage," to reflect the expansion of the onsite spent fuel storage capacity at CPS Unit 1.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosures: 1. Amendment No. 170 to NPF-62
2. Safety Evaluation

cc w/encls: See next page

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Clinton Power Station, Unit 1

cc:

Senior Vice President - Nuclear Services
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President Operations Support
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Licensing and
Regulatory Affairs
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Manager Licensing - Clinton and LaSalle
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Regulatory Assurance Manager - Clinton
AmerGen Energy Company, LLC
Clinton Power Station
RR3, Box 228
Clinton, IL 61727-9351

Director Licensing
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Document Control Desk-Licensing
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Site Vice President - Clinton Power Station
AmerGen Energy Company, LLC
Clinton Power Station
RR 3, Box 228
Clinton, IL 61727-9351

Clinton Power Station Plant Manager
AmerGen Energy Company, LLC
Clinton Power Station
RR 3, Box 228
Clinton, IL 61727-9351

Resident Inspector
U.S. Nuclear Regulatory Commission
RR #3, Box 229A
Clinton, IL 61727

Chief Operating Officer
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Senior Counsel, Nuclear
AmerGen Energy Company, LLC
4300 Winfield Road
Warrenville, IL 60555

R. T. Hill
Licensing Services Manager
General Electric Company
175 Curtner Avenue, M/C 481
San Jose, CA 95125

Chairman of DeWitt County
c/o County Clerk's Office
DeWitt County Courthouse
Clinton, IL 61727

J. W. Blattner
Project Manager
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, IL 60603

Illinois Emergency Management
Agency
Division of Disaster Assistance &
Preparedness
110 East Adams Street
Springfield, IL 62701-1109

AMERGEN ENERGY COMPANY, LLC

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. NPF-62

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by AmerGen Energy Company, LLC (the licensee), dated August 18, 2004, as supplemented on May 13 and 25, June 14, August 17, and October 24 and 25, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications contained in Appendix B, as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-62 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. are hereby incorporated into this license. AmerGen Energy Company, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L.Raghavan for G.Suh/

Gene Y. Suh, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 31, 2005

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 170</u>

<u>FACILITY OPERATING LICENSE NO. NPF-62</u>

<u>DOCKET NO. 50-461</u>

Replace the following page of the Appendix "A," Technical Specifications, with the attached revised page. The revised page is identified by an amendment number and contains marginal lines indicating the areas of change.

<u>Remove Page</u>	<u>Insert Page</u>
4.0-2	4.0-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. NPF-62

AMERGEN ENERGY COMPANY, LLC

CLINTON POWER STATION, UNIT 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By application dated August 18, 2004, as supplemented on May 13 and 25, June 14, August 17, and October 24 and 25, 2005 (References 1 through 7) AmerGen Energy Company, LLC (AmerGen), the licensee, requested a revision to Technical Specification (TS) 4.3, "Fuel Storage," to reflect the addition of fuel storage capacity in the fuel cask storage pool and the increase of fuel storage capacity in the spent fuel pool (SFP) at Clinton Power Station (CPS), Unit 1.

In the supplement dated June 14, 2005, the licensee changed the use of the building crane and the temporary crane. This change may have impacted the staff's original proposed no significant hazards consideration determination published on December 29, 2004 (69 FR 78051). Therefore, a revised no significant hazards consideration determination was published in the Federal Register on August 29, 2005 (70 FR 51093). However, the supplements dated October 24 and 25, 2005, provided additional information that clarified the application, did not expand the scope of the application as noticed on August 29, 2005, and did not change the staff's no significant hazards consideration determination as published on August 29, 2005.

The SFP is currently licensed for 2,512 fuel assembly storage locations. The licensee estimates that the ability to fully offload the core will be lost during the February 2006 scheduled refueling outage, when 312 fuel assemblies are permanently discharged and new fuel is loaded into the SFP for operating cycle 11. The proposed expansion will increase the total storage space from 2,512 to 4,159 fuel assemblies. This additional capacity is expected to allow operation without loss of full-core discharge capability until cycle 15 in the year 2016. The new racks will have a closer assembly-to-assembly spacing to allow for higher density storage and, thus, more storage capability. The new racks will contain Metamic as the active neutron absorbing poison.

In its amendment request, the licensee has proposed to expand the spent fuel storage capacity in two phases. The first consists of adding two new 15 by 12 storage racks in the fuel cask storage pool, providing locations for an additional 360 assemblies and increasing the total capacity to 2,872 fuel assembly storage locations. This will ensure that full-core discharge capability is maintained until February 2008. The second phase consists of removing 12 of the existing racks from the SFP and placing three of the removed racks into the fuel cask storage pool, installing 14 new racks into the SFP and relocating the two racks in the fuel cask storage pool into the SFP.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67, "Accident source term," as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC-19), "Control Room," provides requirements for the design basis accidents radiological consequences.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms."

GDC-62, "Prevention of criticality in fuel storage and handling," requires licensees to prevent criticality in the fuel storage and handling system by physical systems or processes, preferably by use of geometrically safe configurations.

10 CFR 50.68, "Criticality accident requirements," contains requirements for preventing criticality accidents in the SFP.

GDC-61, "Fuel storage and handling and radioactivity control" provides requirements for the residual heat removal capability of the fuel storage systems.

SRP Sections 9.1.2, "Spent Fuel Storage," and 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."

10 CFR Part 20, "Standards for Protection Against Radiation," Subpart B, "Radiation Protection Programs," requires licensees to implement a radiation protection program commensurate with the scope and extent of licensed activities sufficient to ensure compliance with Part 20. Section 20.1101(b) requires licensees to use, to the extent practicable, procedures and engineering controls to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

RG 8.8, Revision 3, "Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Plants Will Be As Low As Is Reasonably Achievable."

RG 8.38, "Control Of Access To High And Very High Radiation Areas In Nuclear Power Plants."

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

3.0 TECHNICAL EVALUATION

3.1 Criticality Considerations

The new spent fuel storage racks are designed to maintain the required subcriticality margin when fully loaded with enriched fuel at a temperature corresponding to the highest reactivity. For reactivity control in the racks, neutron absorber panels will be used. The panels have been sized to sufficiently shadow the active fuel height of all assembly designs stored in the pool. The panels will be held in place and protected against damage by a stainless steel jacket which will be stitch-welded to the cell walls. The panels will be mounted on the exterior or on the interior of the cells in an alternating pattern.

The specified fuel assembly used as the design basis for the racks is a standard GE 10x10 array (GE-14) of BWR fuel rods containing UO_2 clad in Zircaloy, and assumes a uniform average enrichment of 4.8 weight percent U-235. Explicit analyses of all other fuel assembly types were performed to confirm their acceptability for storage in the high-density racks. The effects of the calculations and manufacturing tolerances were evaluated and added in determining the maximum neutron multiplication factor k_{eff} in the storage rack.

A criticality study was performed by Holtec International (Holtec Report No. HI-2033124 in Reference 1), supporting the criticality safety of the new spent fuel storage racks at CPS. The new racks are designed to ensure that the k-effective (k_{eff}) is equal to or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity within the pool (i.e., SFP or fuel cask storage pool) flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations and in mechanical tolerances, statistically combined, giving assurance the true k_{eff} will be less than 0.95 with a 95 percent probability at a 95 percent confidence level. Reactivity effects of abnormal and accident conditions are also evaluated to ensure that under credible abnormal or accident conditions, the reactivity will be maintained less than 0.95 percent. The accidents and malfunctions evaluated included the impact of a dropped fuel assembly on top of a fuel rack; impact on criticality of water temperature and density effects; and impact on criticality of eccentric positioning of fuel assemblies within the rack.

The criticality analyses for the high density fuel storage racks were performed with the CASMO-4 code. CASMO-4 is a two-dimensional multi-group transport theory code. The MCNP code, a three-dimensional transport theory code, and the NITAWL-KENO5a code, a three-dimensional code, were used for verification purposes. The 238-group cross-sections library was used. These methodologies and cross sections are well known and have been accepted in past NRC reviews, including previous analyses by Holtec. The use of the two codes, MCNP and KENO5a provide greater assurance for the analysis accuracy.

The methodologies and cross-sections have been benchmarked, by Holtec (and many other groups) against a number of relevant critical experiments simulating parameters related to storage racks. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k_{eff} calculations for the racks. Holtec has also determined the potential variation of rack and fuel parameters that are used in determining the k_{eff} of the racks and fuel systems. These parameters include rack manufacturing tolerances, poison loading variations, and cell lattice pitch variation. The variation of k_{eff} with these parameters (taken at a 95/95 percent probability/confidence level), was determined.

These parameters were statistically combined with the methodology uncertainty to provide a delta k uncertainty which was added to the base k_{eff} calculation. This treatment of the uncertainties is in conformance with NRC past recommendations and approvals. Therefore, it is acceptable.

Holtec has also investigated abnormal conditions that might be associated with the SFP. These include (1) pool water temperature effects, (2) eccentric fuel positioning (the nominal analysis case with the fuel centered in the cell yields maximum reactivity), (3) dropped fuel assembly (no significant reactivity increase), and (4) rack lateral movement (no significant reactivity increase). These analyses have provided a satisfactory demonstration that reasonably possible abnormal conditions will not lead to a reactivity problem if the required k-infinity and k_{eff} limits are met. The acceptance criterion for safe storage of spent fuel is that the maximum k-infinity be limited to1.33.

Based on the above, the staff concludes that appropriately conservative assumptions were made in the criticality analysis. Therefore, the staff finds the licensee's conclusions acceptable.

3.2 SFP Cooling

The SFP cooling and cleanup system is combined with the upper containment pool cooling system to form one integrated fuel pool cooling and cleanup (FPCC) system. This integrated system is designed to maintain water quality and clarity and to ensure adequate cooling to stored fuel in the SFP, upper containment pool, fuel cask storage pool, and fuel transfer pool. The SFP cooling portion of the system consists of two 100 percent capacity cooling trains each equipped with one pump, one heat exchanger, and associated valves, piping, instrumentation and controls. One train supplies the design basis cooling capacity with cleanup at a rate of 1000 gpm. The second train acts as a backup system and is employed when servicing of the first train is required. The FPCC system cools all of the pools by transferring the decay heat released from the stored spent fuel through the heat exchanger to the component cooling water (CCW) system.

3.2.1 Fuel Pool Heat-Up Analysis

The increase in the number of spent fuel assemblies stored in the fuel pool will be accompanied by an increase in the total decay heat associated with the stored fuel and a corresponding increase in the FPCC cooling loop heat load. The licensee performed thermal-hydraulic analyses to evaluate the effect of the increased storage capacity on the FPCC heat loads and the corresponding FPCC water temperature. Discharge scenarios were considered for both partial and full-core discharges to the SFP and the fuel cask storage pool coupled with only one cooling train operable.

3.2.1.1 Planned Partial Core Offload with a single active failure

The staff has reviewed the licensee’s analysis of the bounding case for the partial core offload. In this scenario it is assumed that 312 assemblies are offloaded to the fuel pool, completely filling all storage locations, beginning 24 hours after reactor shutdown. The minimum decay time for the previously offloaded fuel is assumed to be 18 months. A cooling system configuration that includes a single active SFP cooling system component failure was considered.

In the licensee's evaluation of the maximum SFP bulk water temperature, credit was taken for operation of one FPCC system cooling pump and one FPCC system heat exchanger. The loss of one train of the FPCC system is defined as the single active failure. The licensee used the ORIGEN2 computer code to calculate the decay heat load from previously offloaded fuel, and Holtec's Quality-Assurance validated BULKTEM computer program to calculate the transient decay heat loads and SFP bulk temperatures. An SFP maximum bulk water temperature of 125.8 EF at 87 hours post-shutdown was calculated for this offload scenario with a corresponding coincident net heat load of 27.7 Mbtu/hr. The peak SFP bulk temperature of 125.8 EF for this scenario is below the acceptance criteria of 140 EF. The staff has reviewed the licensee's analysis of the maximum bulk water temperature in the SFP. The projected peak temperature is below the acceptance criteria limit, and is, therefore, acceptable.

3.2.1.2 Full-Core Offload with a single active failure

The staff has reviewed the licensee's analysis of the bounding case for the full-core offload. In this scenario it is assumed that 624 assemblies are offloaded to the SFP starting at 24 hours after reactor shutdown, completely filling all storage locations. The SFP is assumed to be cooled with one FPCC system train operating with CCW at 105 EF. The licensee assumed that the offload began 24 hours post-shutdown and that the fuel is offloaded at a rate of 7 assemblies per hour for the first 160 assemblies followed by 4 assemblies per hour for the remainder of the offload. This offload-rate assumes completion of the full-core offload at approximately 163 hours post-reactor shutdown. The current licensing basis analysis is based on a constant offload rate of 6 assemblies per hour, resulting in the core offload being completed at 128 hours post-reactor shutdown. The licensee used the ORIGEN2 computer code for decay heat calculations. The licensee evaluated the SFP maximum bulk water temperature for this case, incorporating into the analysis the longer overall time to offload the core, and a more precise treatment of the decay heat generated by the spent fuel by using ORIGEN2 calculations. The results of the new analysis showed that the current licensing basis SFP maximum bulk water temperature of 140 EF bounds the newly calculated temperature for the expanded fuel pool. The staff has reviewed the licensee's submittal and finds the heat load reduction obtained by changing the offload rate and using a more precise code for heat load calculation is acceptable and the resulting peak pool temperature remains bounded by that given in the current licensing basis.

3.2.2 Effects of Spent Fuel Pool Boiling

In the unlikely event there is a complete loss of cooling, the SFP bulk water temperature will begin to rise and will eventually reach the boiling temperature. If there were a reduction of SFP water inventory due to boiling, fuel stored in the SFP might become uncovered, leading to fuel failure and a release of radioactivity. The staff has reviewed the licensee's analysis of the consequences of a loss of SFP cooling, which is discussed in the following paragraph.

The licensee has performed analyses to demonstrate minimum time-to-boil and the maximum boil-off rate. Two separate scenarios involving the loss of forced cooling in the SFP were evaluated. The first scenario assumed a partial core offload, and the second scenario assumed a full-core offload. The full-core offload scenario is the limiting case. The calculated minimum time from loss-of-pool cooling at peak SFP temperature until the pool boils based on the heat load for the full core offload scenario is 3.25 hours. The corresponding maximum boil- off rate is 85.9 gpm. The licensee has adequate time to align and supply sufficient water, from a

variety of sources, to the SFP prior to the time to boil. Makeup sources include the normal makeup water system (230 gpm) and shutdown service water system (100 gpm) from each division. Based on the above discussion, the staff concludes that makeup capabilities are adequately provided to mitigate the loss of fuel pool cooling, and the operators have adequate time to respond to the event before boiling could occur. Therefore, the staff finds that the time-to-boil analysis is acceptable

3.3 Materials Compatibility

3.3.1 Structural Materials

The new storage racks to be installed in the SFP are manufactured by Holtec. The storage racks are freestanding and self-supporting, designed to the stress limits of, and analyzed in accordance with, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section III, Subsection NF and NUREG-0800. The material procurement, analysis, fabrication and installation of the rack modules conform to 10 CFR Part 50, Appendix B.

The structural materials used in the fabrication of the new spent fuel storage racks include: ASTM A240-304L for all sheet metal stock, baseplate, and internally threaded support legs, ASTM A564-630 precipitation hardened stainless steel (heat treated to 1100 EF) for externally threaded support spindles, and austenitic stainless steel for weld material.

All of these materials have been previously used in similar applications and are compatible with the spent fuel assemblies. In addition, these materials have a proven history in the SFP environment. Therefore, the staff finds that the use of these materials at CPS is acceptable.

3.3.2 Neutron Absorbing Material

Metamic, a cermat, is composed primarily of B_4C and aluminum (Al 6061). B_4C is the main constituent in materials known to perform effectively as neutron absorbers and Al 6061 is a marine qualified material known for its resistance to corrosion. In spite of these corrosion resistant properties, Metamic has not been previously used in SFP applications.

In its submittal dated October 24, 2005 (Reference 6), the licensee provided a Metamic coupon sampling program, which consists primarily of monitoring the physical properties of the absorber material and performing periodic neutron attenuation testing to confirm the physical properties.

The licensee stated that it will install a coupon tree that holds 10 coupons in the SFP. Each coupon is 8-inches long, 4-inches wide, and 0.075-inches thick. Each coupon will be enclosed in a stainless steel jacket that is representative of the sheathing used on the Metamic® in the spent fuel storage racks and then mounted on the coupon tree. The coupon samples contain 25 percent B_4C, which is consistent with the B_4C content in the Metamic used in the new spent fuel racks.

The coupon tree will be placed in the SFP at a location that will ensure a representative gamma dose to the coupons. The coupons will be removed from the SFP for testing after 2, 4, 8, 12, 20, 28, and 36 years for physical observations and/or confirmatory testing.

The licensee stated that when a coupon is removed in accordance with the sampling program, the following measurements will be performed:

1. Visual observation and photography:

    a. The licensee will observe for visual indications such as bubbling, blistering, cracking or flaking.
    b. Photographs of both sides of exposed coupon will be taken to document coupon condition.

2. Dimensional measurements:

    a. Length
    b. Width
    c. Thickness

3. Weight

The licensee's acceptance criteria are as follow:

- Any change in the length and width of +/- 0.125 inches
- Any change in the thickness of +/- 0.007 inches
- Any change in the weight of +/- 5%

Prior to installing the coupons in the SFP, each coupon is pre-characterized. The physical characteristics presented above are documented for each coupon and included in the documentation package for the coupon tree. These measurements are taken before the coupons have been exposed to any radiation or SFP environment. These measurements define the baseline for the coupons.

When a coupon is removed, measurements and physical observations will be recorded and evaluated for any physical or visual change when compared to the baseline data. If the measurements taken do not meet the established acceptance criteria, the licensee will perform neutron attenuation testing. After all testing is finished, the coupons will be returned to the coupon tree to support the long term testing program.

In order to confirm the validity of the correlation between the physical characteristics of the Metamic coupons and their neutron absorption capabilities, the licensee will perform confirmatory neutron attenuation testing after 4, 12 and 20 years. The testing will be performed regardless of whether the physical measurements taken at these intervals are within the allowable tolerances or not.

To establish a baseline for comparison, the licensee stated that for the areal density for each Metamic coupon taken from the panel material, it will use the fabrication features, qualification and acceptance testing values for the Boron-10 areal density provided by the manufacturer. The manufacturer provides a guaranteed minimum value or a given value with confidence range. The licensee will review the fabrication qualification and acceptance test data to determine whether the value provided is acceptable for pre-characterization. If this value is determined not to be acceptable, the licensee may decide to perform neutron attenuation testing of a coupon prior to installation in the SFP to establish a baseline.

The licensee stated that its coupon program will regularly monitor the Metamic coupon's physical characteristics the same as 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.

Based on its review of the licensee's coupon sampling program, the staff concludes that the materials in the CPS spent fuel racks, including the Metamic® neutron absorber, are compatible with the environment of the SFP. Also, the staff finds the proposed surveillance program, which includes visual, physical and confirmatory tests, is capable of detecting potential degradation of the Metamic® material that could impair its neutron absorption capability. Therefore, the staff concludes that the use of Metamic as a neutron absorber panel in the new spent fuel racks is acceptable.

3.4 Structural Evaluation

The licensee has implemented a complete reevaluation of the mechanical and civil structures to address the structural issues resulting from the expansion. The reevaluation considered the loads from seismic, thermal, and mechanical forces to determine the margin of safety and the structural integrity of the fuel racks, the SFP, fuel cask storage pool and their liners. The loads, load combinations, and acceptance criteria used in the analyses were based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section III, Subsection NF and on NUREG-0800.

3.4.1 Analysis Methodology

The Holtec proprietary software code, DYNARACK, was used for all of the single rack and whole pool multi-rack (WPMR) evaluations. The current version of DYNARACK uses the same algorithm solvers and fluid coupling formulations as the versions used to support previous re-racking requests. There have been no recent key engineering analysis methodology improvements required or performed for the DYNARACK solver. Therefore, the staff finds that the licensee's use of the DYNARACK code is valid for evaluations related to the single rack and WPMR evaluations related to this action.

WPMR analysis for spent fuel racks has been developed and improved by Holtec over a period of many years and is used for determination of global displacements of spent fuel racks, as well as rack-to-rack and rack-to-floor reaction forces. The methodology has been continually improved to minimize the effect of engineering variabilities. Each rack is modeled as a beam-like structure with twelve degrees of freedom. The formulation is set up so that classical beam theory (including shear deformation) static solutions are reproduced when the rack model is subject to end loadings. Therefore, to the extent that the "beam properties" of the spent fuel rack are modeled accurately, the predicted results are consistent with the accuracy of beam theory applied to any structure. The spent fuel rack is a rugged, nearly rigid honeycomb structure whose beam properties (i.e., area and moments of inertia) can be developed with minimal assumptions. Therefore, the results that are obtained from any simulation will differ from the results obtained by considering the racks as rigid, with known mass and inertia properties, only to the extent of the improvements included in the characterization of the beam-like deformations. Based on numerous convergence studies that have been performed over the years, there is confidence that the results obtained reflect the reality of the scenarios under study. The licensee indicated that WPMR performance is noticeably different when the racks

are under water. Therefore, it has been recognized that the responses obtained are very dependent upon the approach used to simulate the hydrodynamic coupling between racks and SFP walls, and between individual fuel assemblies and the cell walls.

Based on the theoretical development, experimental verification, and numerous independent reviews of Holtec's methodology, the licensee asserted that there is a reasonable confidence that the analysis results are within what would be accepted as good engineering accuracy, and any variation is likely to be conservative (i.e., over-predictions).

The licensee performed both WPMR and single rack seismic analyses for CPS's expanded racks. These analyses were based on the simulation of the safe shutdown earthquake (SSE) and the operating basis earthquake in accordance with SRP Section 3.7.1, "Seismic Design Parameters," guidance. The results indicated that the maximum seismic displacements do not pose any potential of impact between the top of the racks and the pool walls. The resultant member and weld stresses in the racks are all below the allowable stresses, with a safety factor of at least 1.2 with respect to the SSE allowable stress. The licensee determined that the minimum safety factor for the cell membrane material and associated welding is 1.22 with respect to the applicable allowable stress and asserted that the racks will remain functional during and after an SSE.

The licensee states that the new rack modules will be separated from each other by a gap of approximately 1.0 inch, and along the pool walls a nominal gap of 2.5 inches will be provided. The licensee indicated that all rack periphery gaps were configured to maximize the number of storage locations that can safely be placed in the pools. The gap sizes between racks and the pool walls were selected to provide gaps that are sufficiently small to prevent inadvertent insertion of an assembly into the gap space during fuel shuffle, and also large enough to limit potential rack impacts to acceptable levels. The licensee also indicated that the 1.0 inch gap between adjacent storage racks was selected as a nominally small value that allows placement of adjacent empty racks without damage. Gaps between racks will be maintained by using half-inch plate extensions beyond the perimeter of the rack storage cell boundary. These base plate gaps will also be maintained at the top by ensuring that the racks are leveled during installation. The installation procedures require that the pedestal extensions be adjusted prior to the rack being inserted into the pool to account for pool floor and bearing pad elevation irregularities. Upon installation, the rack-to-rack and rack-to-wall gaps, as applicable, are measured using long handled measuring devices prior to removal of the lifting rig from the rack.

Rack installation practice is to install the racks with the base plates butted together to the extent possible. Fabrication tolerances and straightness of the side of the base plate would preclude rack base plates on adjacent racks from being in contact over their entire width. However, the DYNARACK model considers that these base plates are in initial contact at the onset of the dynamic simulations. This practically ensures that the base plates will produce in-plane impact loads during the dynamic simulation. The top of the racks is modeled with the nominal gaps. This difference in gaps between the top and bottom of the racks is accomplished in the DYNARACK model by adjusting the length of the rack periphery springs.

Holtec will install the new racks and will provide receipt inspection and pre-installation drag testing procedures that will be used to thoroughly inspect each rack for fit, form, and function. If shipping damage is discovered or a storage cell location exceeds the maximum allowable drag force, tools will be available to expand and repair the cell in question.

The licensee stated that the impacts between the fuel assembly and the cell wall are evaluated for effects on the cell wall integrity and fuel integrity. The method for evaluating fuel integrity entails the development of equivalent maximum accelerations for the five individual fuel masses considered in the DYNARACK model by using the maximum predicted impact load of 815 pounds force (lbf). The equivalent acceleration is then compared to the acceleration that the fuel assembly is capable of withstanding. The cell wall integrity evaluation determines the limit load on an equivalent beam strip (the width of a single cell), which is subject to impact loads taken to occur at the two corners of a fuel assembly grid. The collapse load of the panel is then examined as a bending collapse. Based on classical mechanics limit analysis, the deflected shape of the cell wall is taken as a beam with four plastic hinges, one at each end, and one at each of the load application points.

The licensee stated that the 815 lbf impact load is the maximum value of any fuel-to-cell wall impact from the many impact sites in the total array of simulations. The load occurs in one of the existing racks that is being reused in the new storage configuration. This load is considered to be conservative, since it is predicated based on modeling the fuel assemblies in each rack with five masses. However, a more accurate representation of the fuel assembly impact points would be at the actual fuel grid locations, and the number of these locations exceeds the five lumped masses incorporated in the dynamic analysis model. Therefore, the actual impact loads would be expected to be less, since the individual masses associated with each impact site would be lower than those modeled. The factor of safety of the cell wall to withstand the impact load is determined to be more than 3.0. The factor of safety for the maximum fuel-to-cell wall impact in one of the new racks is determined to be 3.3.

Based on its review, the staff concludes that the licensee has provided adequate and acceptable technical justification in support of the seismic design adequacy and structural integrity evaluation presented in its amendment request.

3.4.2 <u>SFP Operating and Maintenance Experience</u>

The licensee provided a discussion of CPS's past operating and maintenance experience with respect to its spent fuel racks and pool structures including unexpected deformations and damages of racks, fuel assemblies, pool liner and aging related degradation of SFP structural concrete and liner elements. The licensee stated that since installation of the spent fuel racks, there have been no adverse operating or maintenance issues associated with the racks. There have been no unexpected deformations of the racks and there have been no instances of damage to the racks or to fuel assemblies as a result of the use of the racks. As part of the initial installation of the spent fuel racks, drag testing was used to identify individual unusable rack locations (i.e., cells). None of these cells identified as unusable have been used to store fuel assemblies. CPS’s fuel assemblies have experienced fuel channel bow as a result of normal exposure in the reactor core. Fuel assemblies with bowed channels have been stored in the SFP racks without incident.

The CPS's SFP liner is intact with no history of leakage. Welded channels located behind the liner welds route any leakage to flow sensors through interconnected drainage paths. These paths are designed to: (1) prevent the uncontrolled loss of contaminated pool water to other relatively cleaner locations within the containment or fuel building, (2) prevent pressure buildup behind the liner plate, and (3) provide liner leak detection and measurement. These drainage paths are designed to permit free gravity drainage to the fuel building equipment drain system.

Any leakage from the SFP liner is monitored by flow switches and is annunciated in the main control room in the event of excessive leakage.

A level instrument on the SFP surge tanks also monitors pool water volume. Surge tank volume is small compared to the volume of the pools; therefore, surge tank level instrumentation is very sensitive to changes in pool volume. The issue of gamma heating of the SFP structural concrete has been evaluated as part of this re-racking request. There has been no observed age related degradation of the SFP structural concrete from gamma heating.

Based on its review, the staff finds that the licensee's response is acceptable.

3.5 Fuel Assembly Drop

The licensee indicated that Holtec has not performed any specific tests or experiments simulating a fuel assembly drop onto a spent fuel rack. However, the computer code LS-DYNA, which is a commercial code developed by Livermore Software Technology Corporation, was validated under Holtec's 10 CFR Part 50, Appendix B, Quality Assurance program through comparisons with documented test cases. Moreover, LS-DYNA has been widely used for many years in the automobile and aerospace industries to simulate dynamic impact problems, and numerous examples exist in published literature demonstrating the program accuracy when compared with actual test results. The licensee stated that based on the Holtec Analysis, the SFP liner strain is 0.09, which is well below the failure strain limit of 0.38. The licensee’s use of LS-DYNA code in executing the shallow and deep drop events analyses is consistent with good engineering practices. Therefore, the staff finds it acceptable.

Figure 7.5.4,"Deep Drop Scenario 2: Maximum Effective Strain - Concrete," shows that the concrete slab of the CPS SFP is projected to experience very limited local damage as shown by the predicted cracks and the effective strain distribution. The results provided in Figure 7.5.4 of Attachment 5, Reference 1, were determined using the computer code LS-DYNA and are entirely independent of the WPMR analysis. The CPS SFP floor is modeled in LS-DYNA using the Winfrith concrete model, which is part of LS-DYNA's built-in material library. This material model has been validated by the United Kingdom Atomic Energy Authority (AEA) against a wide range of impact and blast tests. The concrete compressive strength is based on the CPS minimum design value of 3,500 pounds per square inch. In reality, the compressive strength is likely to be much higher as concrete continues to harden with age. The use of the Winfrith concrete model together with the minimum concrete compressive strength ensures that the LS-DYNA results are conservative.

The LS-DYNA models used for the drop analyses are developed with some conservative simplifications; the emphasis of each model is placed on the local impact region that could be damaged by the dropped object. For the shallow drop case, only the upper portion of the rack cells near the impacted cell wall was modeled for numerical efficiency and conservatism. The rack is represented by 10 local cells extending 48 inches in the vertical direction; the rest of the rack is assumed structurally safe and absorbs no impact energy in order to maximize the damage in the modeled portion of the rack. Accordingly, displacement constraints were imposed on the bottom nodes of the modeled cells and on the boundary nodes laterally connected to the rest of the rack. For conservatism and simplicity, the stored fuel assembly was modeled by two rigid end fittings connected with a beam that has a mass and cross-sectional area equivalent to all fuel rods in a fuel assembly. The fuel rod yield stress was set to be the buckling stress of the rod. The fuel rod failure strain of the stored fuel assembly was

assumed to be two percent to minimize the resistance of the stored fuel assemblies in a drop event. The dropped fuel assembly was modeled the same way as the stored fuel assembly, except that the top fitting included the weight of the handling tool and the fuel rod material is assumed to be elastic so that the entire dropped mass can be maintained during the drop accident.

The licensee considered only the lower portion of the rack cells near the impact region in the finite element model for deep drop, scenario 1; the rest of the rack is considered structurally safe. The model spans about 65 inches in the vertical direction. Displacement constraints are imposed on the boundary cell wall nodes that are sufficiently far from the dropped fuel assembly. Since the impact is assumed to occur inside an interior cell at the rack center for predicting the maximum base plate deflection, the model takes advantage of the symmetry of the problem by considering only a quarter of the affected structural members with appropriate symmetric boundary conditions imposed. The fuel assembly is conservatively assumed to be rigid. This deep drop model is conservative since the impact energy absorption is limited to the modeled portion of the rack. The model for deep drop, scenario 2, emphasizes the structural responses of the rack pedestal, the bearing pad, the SFP pool liner and the underlying concrete slab. The rack cells are assumed to be safe in this scenario and are not modeled for numerical efficiency because the impact takes place directly above a rack support pedestal. Therefore, the rack is represented only by a pedestal cylinder and the mating female block, as well as a portion of the rack base plate with mass density adjusted to match 1/4 of the total weight of the smallest empty rack. The model also includes the bearing pad, the stainless steel liner, and the concrete slab resting on subgrade. The concrete slab model is fixed at the bottom surface and restrained from moving in the horizontal directions at the periphery to simulate the confining effect by surrounding concrete. Finally, the fuel assembly is represented by a rigid fuel assembly end fitting modeled with the full dropped weight.

Based on its review, the staff finds that the licensee's methodology and conclusions are acceptable.

3.6 Fuel Handling Accident

The licensee evaluated the impact of newly installed higher density storage racks in the SFP and fuel storage in the fuel cask storage pool on the current design basis accidents (DBA) dose analyses, as discussed in the CPS USAR. The DBAs that are potentially affected by the proposed change to the spent fuel storage capacity are the fuel handling accident (FHA) and the cask drop accident. By Amendment No. 147, dated April 3, 2002, the CPS licensing basis for the FHA was changed by a selective implementation of an alternative source term, per the provisions of 10 CFR 50.67. In support of that amendment request, the licensee demonstrated that the radiological consequences of an FHA, either in the containment or in the fuel building, are within the offsite and control room dose acceptance criteria specified in SRP Section 15.0.1 and GDC-19, and well within the dose criteria given in 10 CFR 50.67.

The NRC staff has reviewed the licensee’s analysis of the proposed action on DBA dose analyses. The licensee stated that the radiological consequences of accidents in the fuel building or containment are not affected by the expansion of spent fuel storage capacity. The staff evaluated this statement with respect to the radiological consequences of DBAs for operation after the new fuel storage racks have been installed and also during the installation of the racks in the SFP and fuel cask storage pool.

With respect to the postulated FHA after the new fuel storage racks are installed, the licensee did not propose changes to the procedures and equipment used to move fuel; therefore, there would be no change to the projected fuel damage. Additionally, the licensee did not propose changes to the fuel burnup, decay time or pool water level; therefore, the postulated radiological source term would remain the same. Adding additional spent fuel storage does not increase the amount of fuel assumed to be damaged in an FHA, and the proposed action does not significantly change the source term in the DBA. Therefore, the staff finds that the current licensing basis FHA dose analysis remains applicable after the expansion of the spent fuel storage capacity.

The current licensing basis analysis of the cask drop accident is discussed in the CPS USAR, Chapter 15.7.5. This accident postulated that a fuel shipping cask falls into the fuel cask storage pool as a consequence of an unspecified failure of the cask lifting mechanism. The cask cannot travel over the spent fuel storage pool because of physical barriers and a redundant limit switch on the fuel building crane; therefore, no spent fuel in the spent fuel storage pool can be damaged by the cask drop. In the CPS USAR evaluation, the licensee stated that there are no radiological consequences associated with a cask drop accident because there is no postulated release of radioactive materials. With the subject amendment request, the licensee plans to install spent fuel storage racks in the fuel cask storage pool. Spent fuel would be stored in the fuel cask storage pool, where previously it had not been stored. The licensee states that they plan to implement administrative controls to ensure that fuel will be removed from the racks in the fuel cask storage pool prior to any fuel cask being moved in the area. Therefore, there would be no damage to spent fuel or radiological consequences as a result of a cask drop on the empty fuel storage racks in the fuel cask storage area. Therefore, the staff finds that the current licensing basis analysis of the cask drop accident remains bounding, with respect to radiological consequences.

During removal and installation of fuel storage racks in the SFP and fuel cask storage pool, the licensee will ensure that all work in the SFP and fuel cask storage pool area will be controlled and performed in strict accordance with specific written guidance. Any movement of fuel assemblies required to support removal and installation of racks will be performed as during normal refueling operations, and no shipping cask movement will be performed during this time frame. The licensee will determine and follow safe load paths and written procedures to ensure that no racks are carried over any portions of the existing fuel storage racks containing fuel assemblies. With the proposed limitations on rack and cask movement, there would be no release of radioactive material from damaged spent fuel and no radiological consequences due to fuel rack installation.

The licensee has also evaluated the impact of expanded fuel storage on the consequences of a loss of SFP cooling. The licensee has determined that in the unlikely event of a total loss of SFP cooling, there would be sufficient time for the operators to provide alternate means of cooling to preclude fuel damage and radioactive material release. Therefore, there are no projected radiological consequences due to loss of SFP cooling.

The staff finds that the current DBA dose analyses remain bounding for the installation of expanded spent fuel storage capacity in the SFP and fuel cask storage pool, and are, therefore, acceptable.

3.7 Load Paths and Heavy Load Analysis

NUREG-612 provides guidelines for licensees to ensure safe handling of heavy loads by prohibiting load travel, to the extent practicable, over spent fuel assemblies, over the core, and over safety-related equipment.

The fuel building crane is a single-failure-proof crane with sufficient capacity to handle all lifts during the reracking process. The crane will be used to lift new and existing fuel racks between the truck bay and the operating deck to enable access and egress from the building. However, because of physical travel limits of the fuel building crane, which prevent use of the main hook over the east end of the SFP, the fuel building crane cannot be used to install and remove all of the racks within the SFP. The licensee has proposed to use a temporary crane to install and remove racks along the east end of the SFP. A temporary crane will be constructed and installed on the fuel handling platform rails to gain access to the affected racks. The temporary crane will be configured such that it may also be used to access all of the racks in the SFP. The licensee will consider the overhead crane as a backup system to the newly reconfigured full height temporary crane. The design of the temporary crane includes personnel platforms supported by the end trucks of the crane. These platforms will move along with the crane and will only be in place while the temporary crane is in place on the fuel handling platform rails.

The licensee states, in attachment 6 of Reference 1, that the safe handling of heavy loads by both the fuel building and the temporary crane will be ensured by following the "defense-in-depth" approach guidelines of NUREG-0612. In its implementation of these guidelines the licensee states that it will do the following: (1) use defined safe load paths in accordance with approved procedures, (2) require supervision of heavy load lifts by designated individuals responsible for ensuring procedural compliance and safe lifting practices, (3) train all crew members involved in the use of the lifting and upending equipment, using programs that satisfy the requirements of ANSI/ASME 30.2 - 1976, (4) use lifting devices (slings) that are selected, inspected and maintained in accordance with ANSI B30.9 - 1971, (5) perform inspection, testing, and maintenance of cranes in accordance with ANSI/ASME B30.2 -1976, (6) ensure the designs of the fuel building crane and the temporary crane meet the requirements of CMAA-70 and ANSI/ASME 30.2 -1976, and (7) ensure reliability of special lifting devices by application of design safety margins, and periodic inspections and examinations using approved procedures. The NRC staff has reviewed this approach and found that this approach fully satisfies the criteria of section 5.1.1 of NUREG-0612 and is acceptable.

In a letter dated June 14, 2005 (Reference 4), the licensee indicated that the temporary crane will be used to lower racks into the pool, lift racks out of the pool, and move the racks horizontally in the pool at a limited lift height above the pool floor. New racks to be installed in the SFP will be lowered by the temporary crane to a point approximately 6 inches above the SFP floor. Once the new rack is lowered to a minimal height above the SFP floor, any further movement of the rack will be performed at this minimal height. Existing racks will be lifted from the SFP floor no higher than approximately 6 inches and will travel at this height until reaching the point of egress from the SFP. At this point, the existing rack will be lifted to the surface of the SFP.

Since the temporary gantry crane is not single-failure proof, the potential of a heavy load drop must be considered. The licensee evaluated a postulated rack drop accident and concluded that the accident will not result in significant damage to the pool floor liner or in a loss of pool water. The licensee also indicated that the fuel building crane will only be used as a

contingency in the event the temporary crane is inoperable. Since the licensee's analyses for a rack drop from the temporary crane demonstrate acceptable consequences, and the fuel building crane is single-failure-proof, the staff finds the use of these cranes for rack installation to be acceptable. In addition, specific procedures will be in place to govern all rack movement activities. Existing plant procedures will continue to be used to operate the fuel building overhead crane. A safe load path will be prepared for the crane movement and evaluated by site engineering staff. Barriers will be in place to ensure the safe load path is used and that no heavy loads travel over fuel.

The staff finds the licensee has provided adequate assurance that its planned actions for the handling of heavy loads for the installation of the storage racks are consistent with the "defense-in-depth" approach to safety described in NUREG-0612. The staff finds that the use of the fuel building crane and temporary gantry crane is acceptable because it will enable the licensee to maintain safety during the handling of heavy loads associated with the SFP expansion.

3.8 <u>Occupational Radiation Exposure</u>

The licensee plans to replace the existing fuel storage racks in the SFP with 16 new high-density racks, and permanently install three of the existing racks in the fuel cask storage pool, in a two-phase operation. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the fuel rack installation can be performed for between 4 and 14 person-rem. This estimate includes the radioactive waste processing of the existing contaminated racks, as well as the projected dose to divers.

All operations involved in the fuel rack installations will follow detailed procedures prepared in accordance with ALARA principles. Personnel performing the re-racking operation will be given pre-job briefings to ensure awareness of job responsibilities and necessary precautions. Radiation protection personnel at CPS will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and assure that exposures are maintained ALARA. Personnel monitoring equipment (including TLDs), protective clothing, and respiratory protective equipment will be issued as required. Alarming dosimeters will be used as needed to confirm exposure and dose rates to workers.

The licensee plans to use divers in the pool to remove underwater interferences and assist in fuel storage rack removal. Procedures for controlling diving operations will comply with the guidance in RG 8.38. During the diving operations, the licensee estimates that dose rates will average from 20 to 40 mrem/hr. Special precautions such as physical barriers or tethers will be used to prevent a diver from coming in close proximity to highly radioactive materials in the pool. The diver will be confined to a safe diving area within the pool, which will be clearly delineated in the pre-job brief as well as physically marked in the pool.

The diver will be visually monitored, either directly or remotely, at all times during the dive. In addition, the diver will be monitored by a remote dose telemetry system. This system enables the radiation protection personnel supervising the dive to obtain the dose being delivered to the diver's body and extremities. The diver will have a hand-held probe to complete radiological surveys when entering the water and when changing locations within the pool.

The divers will be monitored for radiation and contamination, as will all items removed from the pool. Appropriate measures will be taken to minimize the spread of contamination. The existing fuel racks that are removed from the SFP will be rinsed and surveyed as they break the water's surface, allowed to 'drip dry', and then placed in plastic shipping bags to contain any contamination until they are placed in shipping containers to be taken offsite for disposal.

The increased storage capacity will not affect dose rates in areas adjacent to the SFP and transfer canal. The concrete side walls of the SFP provide sufficient shielding such that the maximum dose rate in adjacent areas from fuel in the SFP is calculated to be two mrem/hr, if the pool is completely filled with freshly offloaded fuel. The walls of the fuel cask storage pool are not as thick, and the licensee's shielding calculations indicate that filling the racks that are proposed to be installed in the fuel cask storage pool with freshly offloaded fuel could result in dose rates of up to 26 mrem/hr in adjacent areas. This could be mitigated by filling the outer (peripheral) three rows of the storage cell with older (more decayed) fuel, thus reducing the maximum dose rate in the adjacent areas to 4.4 mrem/hr. The licensee will implement administrative controls to ensure that fuel stored in the peripheral storage cells will have been stored outside of the reactor for a minimum of 10 years, allowing sufficient decay time.

The NRC staff concludes that the CPS SFP re-racking operations can be performed in a manner that will ensure that doses to workers will be maintained ALARA. The staff concludes that the projected dose for the project of 7 to 14 person-rem is in the range of doses for similar modifications at other nuclear plants, and is, therefore, acceptable.

3.9 Solid Radioactive Waste

The existing contaminated fuel storage racks will be the main source of radioactive waste for the re-racking. The old existing fuel storage racks will be washed down prior to being removed from the pool to remove as much contamination as possible. Then the racks will be shipped to a volume reduction facility for processing and subsequent disposal at a burial site. Shipping containers and procedures will conform to Federal regulations as specified in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," and to the requirements of any state through which the shipment may pass, as set forth by the state department of transportation.

Spent resins are generated by the processing of SFP water through the pools' purification system. These spent resins are disposed of as solid radioactive waste. Resin replacement is determined primarily by the requirement for water clarity and is normally done approximately once per year. No significant increase in the volume of solid radioactive waste is expected with the expanded storage capacity. During pool re-racking operations, small amounts of additional waste resin may be generated by the pools' cleanup systems on a one-time basis. The licensee stated that the generation of additional solid radioactive waste will be minimized during the re-racking operation. The staff finds the licensee's statement acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comments on such finding (69 FR 78051 and 70 FR 51093).

Pursuant to 10 CFR 51.21, 51.32, and 51.35 an Environmental Assessment and Finding of No Significant Impact has previously been prepared and published in the *Federal Register* on October 25, 2005 (70 FR 61651).

Accordingly, based on the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant impact upon the quality of the human environment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Request for Technical Specification Change to Support Onsite Spent Fuel Storage Expansion," dated August 18, 2004.

2. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Additional Information Supporting the Request for License Amendment Related to Onsite Spent Fuel Storage Expansion," dated May 13, 2005.

3. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Affidavit for Withholding of Global Nuclear Fuel Proprietary Information in Support of Onsite Spent Fuel Storage Expansion," dated May 25, 2005.

4. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Additional Information Supporting the Request for License Amendment Related to Onsite Spent Fuel Storage Expansion," dated June 14, 2005.

5. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Revised No Significant Hazards Consideration Supporting the Request for License Amendment Related to Onsite Spent Fuel Storage Expansion," dated August 17, 2005.

6. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Additional Information Supporting the Request for License Amendment Related to Onsite Spent Fuel Storage Expansion," dated October 24, 2005.

7. Letter from Keith R. Jury, Licensing and Regulatory Affairs, AmerGen Energy Company, LLC, to U.S. Nuclear Regulatory Commission, "Correction to Additional Information Supporting the Request for License Amendment Related to Onsite Spent Fuel Storage Expansion," dated October 25, 2005.

Principal Contributors: Anthony Attard
Yamir Diaz
Michelle Hart
David Jeng
Angelo Stubbs
Roger Pedersen

Date: October 31, 2005