CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-101
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United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE STEAM GENERATOR REPLACEMENT
AND POWER UPRATE LICENSE AMENDMENT APPLICATIONS

Dear Sir or Madam:

By letters dated October 4, 2000 and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). NRC letter dated June 18, 2001 requested additional information to support staff review of the proposed license amendment requests. The requested information is provided by the Enclosure to this letter.

The enclosed information is provided as a supplement to our October 4, 2000 and December 14, 2000 submittals and does not change the purpose or scope of the submittals, nor does it affect the conclusions of either the no significant hazards considerations or environmental evaluations previously submitted.

Please refer any questions regarding the enclosed information to Mr. Mark Ellington at (919) 362-2057.

A001

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Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough
Notary (Seal)

DARLENE S. YARBOROUGH
NOTARY
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PUBLIC
HARNETT COUNTY, NC

My commission Expires: 2-21-2005

KWS/kws

Enclosure

c: Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. N. Kalyanam, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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NRC Question 1

In Section 2.3.2 of Reference 1, you indicated that the computer code used for the pipe stress analysis is different from that used in the original design basis analysis. Provide a justification that the new code was benchmarked for this application.

CP&L Response

As indicated in Section 2.3.2 of Reference 1, the *original* analysis for the Harris Nuclear Plant (HNP) piping was performed using the PIPESTRESS 2010 computer code. Subsequent to the issuance of the HNP operating license, however, CP&L adopted the ADLPIPE computer code for performing pipe stress analysis. Benchmarking of the ADLPIPE computer program is procedurally controlled by CP&L engineering analysis software dedication and benchmark requirements. Thus, the use of ADLPIPE is not a change with respect to the pipe stress analyses performed for the HNP SGR/uprate.

NRC Question 2

In Section 2.16.1.2-3 of Reference 1, you stated that the reactor coolant system (RCS) support loads on the internal concrete structures affect primarily the localized support areas. However, you did not discuss your evaluation of the local areas for increased pipe support loads. Provide a summary of the evaluation of local support areas for increased RCS support loads. If an evaluation was not performed, provide the basis for its exclusion.

CP&L Response

Loads on some RCS supports changed as a result of SGR/uprate considerations. These load changes were specifically evaluated as to their impact, including the impact on localized areas. The results of these evaluations, which are documented in the Addenda to the existing design basis calculations, indicated that all of the changes were well within the acceptable values, and the supports are structurally adequate to safety resist the changed loadings.

NRC Question 3

In Section 2.16.1.2-4 of Reference 1, in evaluating three main steel platforms you stated that for qualifying internal steel structural platforms, base temperature of 60 ^{0}F was used, although the effective base temperature is higher than 60 ^{0}F. Provide the magnitude and justification for the higher effective base temperature used as a basis for qualifying internal steel structural platforms.

CP&L Response

The temperature differential effects were considered for the main steel platforms in the following manner. The construction stage temperature of 60 ^{0}F and the normal operating temperature of 120 ^{0}F were used for the normal operating condition. The effect of initial ΔT of 60 ^{0}F (120-60), on the thermal loads reduces significantly as both concrete and steel are exposed to the 120 ^{0}F temperature for long duration and as both expand. The effect of the accident temperature on the main steel platforms is based on the maximum accident temperature minus the 120 ^{0}F and plus the effective residual initial ΔT effect from long term exposure to normal operating temperature. Considering the long term exposure to 120 ^{0}F, the equivalent base metal temperature was calculated to be 110.77 ^{0}F. This value was conservatively rounded off to 105 ^{0}F in the design. This same base metal temperature was also used in the evaluations of thermal effects with SGR/uprate considerations.

NRC Question 4

In Section 2.16.1.2-4 of Reference 1, with regard to the main steel platforms at elevation (EL) 236', 261', and 286', you stated that the governing load cases for the majority of critical steel member/connections do not include temperature load because these structures are generally free to expand under thermal loads due to slotted holes. Describe the method(s) you used to ensure that steel member/connections that are not free to expand under thermal loads have been evaluated for increased thermal loads in combination with the other design basis loads

CP&L Response

All critical members/connections of the main steel platforms identified in the existing final design basis calculation were evaluated for the effects of the increase of DBA temperature due to the SGR/uprate consideration. Critical members/connections with both slotted and regular connections were included in these evaluations for all applicable loads and load combinations. The evaluation results, which are documented in an Addendum to the existing design basis calculation, indicated that the main steel platforms are structurally adequate for the increase in DBA temperature.

NRC Question 5

In Section 2.16.1.2-4 of Reference 1, with regard to the main steel platforms at EL. 236', 261', and 286', you stated that higher allowable stresses are allowed for cases that include the accident temperature, and concluded that there is sufficient margin available to accommodate the potential increase of accident temperature in the main steel platforms due to the steam generator replacement/power uprate (SGR/Uprate). Provide the design basis margin and margins after considering increased accident temperature loads due to the SGR/Uprate.

CP&L Response

The impact of the DBA temperature on the main steel platforms due to the SGR/uprate consideration was evaluated based on the existing design basis calculation. The results, which are documented in an Addendum to the existing calculation, demonstrated that the effective DBA temperature increase is only approximately 2 %. The effect of this temperature increase on the overall existing design margin, which is calculated for load combinations that include contribution of other loads in addition to the temperature loads is insignificant and the original margins are not affected by the SGR/uprate consideration.

NRC Question 6

In Section 2.16.1.2-4 of Reference 1, for main steam tower and main steam/feedwater hard restraint structures, and steam generator access platforms, provide evaluation summaries and design margins as a result of the power uprate.

CP&L Response

The effects of the revised DBA temperature on the subject steel structures were evaluated based on the existing pertinent design basis calculations. These evaluations were included in the Addenda to the pertinent calculations. The following are the summary of results:

- MS Towers & MA/FW Hard Restraints:
  Minimum design margin of 8 % exists to accommodate contribution from 7.64 % increase in the DBA temperature load on the overall margin, which is based on various loads and their combinations in addition to the temperature loads.

- SG Access Platforms:
  Minimum design margin of 25 % exists to accommodate contribution from 7.64 % increase in the DBA temperature load on the overall margin, which is based on various loads and their combinations in addition to the temperature loads.

NRC Question 7

In Section 2.16.2.2 of Reference 1, you stated that the existing peak accident pressure in the main steam tunnel is 6.47 psig. However, the Pre-SGR/Uprate main steam tunnel accident pressure provided in Table 2.23-2 is 18 psia (i.e., 3.3 psig). Discuss why the existing and Pre-SGR/Uprate main steam tunnel accident pressure is different in section 2.16.2.2 of reference 1 and in Table 2.23-2, respectively. Also, describe how the dynamic effects of the post SGR/Uprate accident pressure time history profile shown in Table 2.23-4 have been considered in the structural evaluation of the main steam tunnel, and internal steel structures and components.

CP&L Response

The design pressure for the main steam tunnel structural design was conservatively established as 16 psig, which is a conservative round off of calculated peak pressure of 6.47 psig multiplied by a dynamic load factor of 2. The peak pressure of 3.3 psig (18 psia) shown in Table 2.23-2 was the final main steam tunnel peak pressure that was bounded by the design pressure.

The accident pressure with SGR/uprate considerations, using the same dynamic load factor of 2, is 10.2 psig, which is less than the design accident pressure of 16 psig. Therefore, there is no impact on the main steam tunnel design.

NRC Question 8

In Section 2.16.2.2 of Reference 1, you stated that the increase in the maximum temperature in the main steam tunnel will affect the platform steel. Explain how the temperature increase in pipe support structural steel members that provide support to the platform steel was considered. You also concluded that structural failure of platform steel will not occur due to increase in main steel tunnel temperature and that the SGR/Uprate does not introduce seismic II/I concerns with these platforms. Provide a basis, including a description of analyses and evaluations, to justify your conclusions.

CP&L Response

The increase in temperature in pipe support structural steel members was considered by performing a heat transfer thermal analysis wherein the thermal profile, based on thermal-hydraulic analysis, was applied to various structural steel members of the support frame to derive the maximum average temperature for the overall support frame. Based on the heat transfer analysis, the maximum average temperature(s) were found to be between 257 ^{0}F to 294 ^{0}F for various support frames for the SGR/uprate conditions. For individual structural steel members, this translated into an increase in the thermal loading between 2.57 % to 7.64 % compared to the existing analysis. The pipe support structural steel was reviewed for temperature increase against the existing design margins. The evaluations are documented in calculations related to the individual supports. All pipe support structural steel members were found to be acceptable for the increase in thermal load.

There are three levels of steel access platforms (walkways) provided in the Main Steam Tunnel. These steel framed platforms, a small one at Elevation 272.50', a larger one at Elevations 280.00' and 282.00', and a third one at Elevation 284.00' are provided for access to and maintenance of various equipment/components. These platforms are primarily supported off the massive pipe support structural steel members and the tunnel walls at the periphery, and they are constructed of lightweight steel members. In the original design basis calculations of these light walkway structures, temperature loads were not specifically considered. However, slotted holes were provided on design drawings and design details to release thermal stresses due to restraint conditions. The temperature related stresses in the platform members and their connections are self-limiting in nature and these temperature stresses are accommodated globally by the slotted holes and the stress redistribution in members and connections. The seismic "II over I" considerations of these relatively light platform structures during and after the postulated accident condition are not considered to be affected by the new temperature values determined by the new evaluations. The platform steel and some connections may experience some localized yielding as a result of this temperature increase, and some platform deformations may take place; however, gross failure is not expected to occur under this temperature increase due to the considerations described above. Therefore, the accident temperature with the SGR/uprate considerations has minimal effect on the overall structural stability and adequacy of these platforms.

NRC Question 9

In Table 1-1 of Reference 2, you stated that for reactor internal components evaluation one of the computer codes is different from those used in the original design basis analysis. Provide a justification that the new code was benchmarked for this application.

CP&L Response

ANSYS is a general-purpose finite element code widely used in the nuclear industry. It is provided with a very substantial verification package to demonstrate appropriate performance and accuracy. It is configuration controlled within Westinghouse. The verification process also includes running the same verification problems on the Westinghouse computer servers to ensure consistency. It should be noted that the type of analyses performed for the HNP uprate do not utilize any special capabilities of the previous WECAN Code. The analysis can be performed equally well using ANSYS. Therefore, the use of ANSYS in the HNP uprate would be considered an appropriate application.

ANSYS has been applied and acceptance granted in the recently completed Westinghouse WOG Baffle Bolting Program. As part of WCAP-15029-P-A, Revision 1, dated December, 1998, titled "Westinghouse Methodology for Evaluating the

Acceptability of Baffle-Former-Barrel Bolting Distribution Under Faulted Load Conditions," the NRC issued a SER titled "SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION WCAP-15029, WESTINGHOUSE METHODOLOGY FOR EVALUATING THE ACCEPTABILITY OF BAFFLE-FORMER-BARREL BOLTING DISTRIBUTIONS UNDER FAULTED LOAD CONDITIONS" (principal contributor: F. Grubelich, NRR/EMEB (301) 415-2784, date November 10, 1998, project No. 694). In this SER, the Commission has given their approval of the methodology to determine acceptable bolt patterns (which includes the utilization of the ANSYS code). Three Westinghouse reactor plants have applied this methodology.

NRC Question 10

In Section 5.1.1.2 of Reference 2, with regard to the evaluation of the core support pads, you stated that the combined normal plus loss-of-coolant accident (LOCA) stresses were compared to the applicable faulted condition acceptance criteria. Provide a justification for not combining stresses due to safe shutdown earthquake with LOCA stresses in the faulted load combination.

CP&L Response

The maximum stress intensities reported in Table 5.1.1.2 of Reference 2 are due to the faulted condition limit load for the core support pads of 2,620,000 lb. not the 2,100,000 lb. LOCA load. According to the current analysis, the 2,100,000 lb. total LOCA load on the core support pads provides sufficient margin under the 2,620,000 lb. faulted condition load for an SSE seismic load of more than 1,000,000 lb. to be combined by square-root-sum-of-the-squares (SRSS). The margin actually allows an SRSS combination with a total horizontal force of 1,566,000 lb. The HNP-specific SSE load at the core support pads was calculated to be only 409,916 lb. Thus, even combining the LOCA and SSE loads by direct addition can be shown acceptable.

NRC Question 11

In Table 5.1.1-1 of Reference 2, you stated that the maximum range of stress intensity for reactor vessel closure studs, 97.5 ksi, is less than the code-allowable stress of 80.1 ksi. This statement appears to be in error. Provide a justification for the adequacy of the reactor vessel closure studs.

CP&L Response

The reactor vessel closure stud information in Table 5.1.1-1 of Reference 2 contains a typographical error. The correct code allowable stress for the reactor vessel closure studs is 110.25 ksi that is $3S_m$ for the SA-540, Class 3 stud material at an operating temperature of 550°F. The data provided in Table 5.1.1-1 of the licensing report should show $3S_m$ =110.25 ksi, not 80.1 ksi. Therefore, the vessel closure studs are acceptable.

NRC Question 12

In Section 5.5.1.3.3 of Reference 2, with regard to the steam generator displacements, discuss how the steam generator displacements were addressed in evaluating the steam generator attached piping.

CP&L Response

The peak values of steam generator displacements (3 translations) and rotations (3 rotations) (supplied by Westinghouse) are multiplied by a dynamic factor and were then utilized in the steam generator attached piping analysis as static load cases. The results from 6 equivalent static cases are combined by absolute summation.

NRC Question 13

In Section 5.6.1.2 of Reference 2, with regard to fatigue analysis of reactor coolant pump, you indicated that none of the changes to the normal or upset transients for the SGR/Uprating cause a non-significant pressure or thermal transient to become a significant transient. Discuss the criteria used in determining the significance of pressure and thermal transients for the fatigue analysis of the reactor coolant pump.

CP&L Response

Westinghouse reviewed the Normal Condition and Upset Condition transients and compared the temperature changes (ΔT) and the pressure changes (ΔP). These are of interest to the stress intensity range and fatigue considerations. In some cases, the SGR/uprating ΔP and/or the ΔT are equal to or less than the current equipment specification values and are thus bounded and require no additional investigation. For cases in which ΔP or ΔT increased, the effect on the various generic stress reports was reviewed.

Since the generic reports use the ASME Code fatigue waiver, only a significant fluctuation is of interest in fatigue. Using 304 stainless steel properties, the Code considers the following pressure and temperature fluctuations as significant enough to include in a fatigue waiver evaluation of NB-3222.4(d).

$$\text{Significant } \Delta P = \frac{P}{3}\left(\frac{S}{S_m}\right) = \frac{2500}{3}\left(\frac{26}{20.0}\right) = 1083 \text{ psi}$$

where,

P = 2500 psi (Design pressure)

S = 26 ksi (S_a at 10^6)

S_m = 20.0 ksi for 304 grades (allowable stress intensity at seal housing operating temperature of 200°F)

and,

$$\text{Significant } \Delta T = \frac{S}{2E\alpha} = \frac{26{,}000}{2(25.7)(10.35)} = 48.9°\text{F}$$

where,

S = 26 ksi (S_a at 10^6)

E = $25.7(10^6)$ psi (Young's Modulus for casing temperature of 557°F)

α = $10.35(10^{-6})$ in/in/°F (Coefficient of thermal expansion for casing temperature of 557°F)

For the SGR/uprating, a review of the changes to the Normal or Upset transients showed that none cause a non-significant pressure or thermal transient to become a significant transient. Thus the fatigue waiver evaluation of NB-3222.4(d), for the current HNP licensing basis/acceptance requirements bounds the SGR/uprating.

NRC Question 14

In Section 5.8 of Reference 2, with regard to the pressurizer, discuss the evaluations performed for the pressurizer safety valves and the power-operated relief valves.

CP&L Response

Westinghouse performed the NSSS Component Sizing evaluation for the pressurizer safety valves (PSVs) and pressurizer PORV evaluations that are covered in Section 4.3.1.3 of Reference 2. The installed capacities of these components were evaluated at the SGR/uprating conditions.

The sizing basis for the PSVs is designed to limit the pressurizer pressure not to exceed 110 percent of the RCS design pressure on a complete loss-of-load transient. This criterion is conservatively met if the total capacity of the pressurizer safety valves is greater than or equal to the peak pressurizer in-surge flow rate during and following this transient.

The sizing basis for the PORVs is to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint for the design basis full-load rejection with steam dump transient. This criterion is conservatively met if the total PORV capacity is greater than or equal to the peak pressurizer in-surge flow rate during and following this transient.

Pressurizer Safety Valves

The total capacity of the installed pressurizer safety valves at set pressure plus 3-percent accumulation is higher than the calculated maximum required capacity at SGR/uprating and SGR/current power conditions. The installed pressurizer safety valves are therefore acceptable at SGR/uprating and SGR/current power conditions.

Westinghouse also states that the adequacy of the installed pressurizer safety valves is contingent upon the safety analyses in Section 6.2, Non-LOCA Transients, meeting their appropriate accident analysis acceptance criteria. The installed pressurizer safety valves are acceptable at SGR/uprating and SGR/current power conditions, since the valves performance was modeled in NSSS Section 6.2 (Chapter 15 Non-LOCA Safety Analysis) and met Chapter 15 acceptance criteria.

Pressurizer PORVs

The total capacity of the installed pressurizer PORVs is higher than the required capacity at SGR/uprating and SGR/current power conditions and is, therefore, acceptable for SGR/uprating and SGR/current power conditions.

NRC Question 15

Confirm that safety-related motor-operated valves (MOVs) will be capable of performing their intended function(s) following the SGR/uprate, including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the SGR/uprated power level was not evaluated.

CP&L Response

HNP MOV calculations are based on preparation of the design ΔP for a specific valve in the MOV program for the required operation. To the extent that fluid flow, temperature, and pressure are important for a specific ΔP calculation, of the design ΔP, they were included. Typically conservative, bounding values are used for fluid density and the maximum Δp occurs at no flow conditions (when a valve reaches the closed position or is opening from the closed position). The changes in ambient temperature were evaluated for impact on the motor operators.

With regard to the second part of the question, it is understood that this request is not specific to MOVs. The response provided below is divided between the NSSS systems and the BOP systems. Rather than provide a specific list of systems and components that were not evaluated, the response describes the process used to identify affected systems and components. Functionality was evaluated on a system level for those systems that were determined to be unaffected.

For NSSS systems, the methodology for determining which mechanical systems are not impacted by the SGR/PUR is as follows: The process parameters at the SGR/uprate conditions (pressure, temperature, etc.) were reviewed to determine whether or not they remain within the range of the Westinghouse analysis of record design parameters. The no-impact systems review process documented why system functional requirements, performance criteria, process parameters, and the critical calculations are not affected. The summary for the affected NSSS systems is presented in the NSSS Licensing Report, Enclosure 6 of the SGR license amendment request (CP&L letter HNP 00-142, dated October 4, 2000).

For BOP systems, the systems were initially screened for SGR/PUR impacts in a "Design Review Activity." Systems were screened out if they met the following criteria:

- The particular system does not interface with other systems that have been affected.
- The system is not subjected to the environmental effects associated with the affected systems (e.g. heat loads, radiation)
- The system is not affected by normal, abnormal or post-accident conditions
- The system is not a physical structure or support system associated with the affected system
- The system does not perform a process function in support of the revised system.

The results of the reviews of potentially affected structures, systems and components are summarized in the BOP Licensing Report, Enclosure 7 to the SGR license amendment request (CP&L letter HNP 00-1142, dated October 4, 2000) and Enclosure 6 of the PUR license amendment request (CP&L letter HNP-00-175, dated December 14, 2000).

NRC Question 16

Clarify whether you have evaluated the effect of increased temperature due to power uprate on thermally induced pressurization of piping runs penetrating the containment that were evaluated in response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions."

CP&L Response

The effect of increased containment temperature on thermally induced pressurization of piping runs penetrating the containment has been evaluated. As explained in our letter Serial HNP-98-097 of 9/28/98 in response to Generic Letter 96-06, only two containment penetrations (M-74 and M-76B) have internal pressures that are affected by containment temperature. The evaluated internal pressures in the other penetrations are controlled by such parameters as valve configuration and design.

For the two affected penetrations, the temperature used in determining the internal pressure for Generic Letter 96-06 was 260 ^{0}F. This temperature was not recalculated specifically for power uprate, but the maximum containment liner temperature was determined to be 255.3 ^{0}F. The maximum penetration piping temperature would not exceed the liner surface temperature because the piping for these two penetrations starts at ambient and would take longer than the liner surface to heat up since the entire pipe and fluid contents must be heated. Thus, the 255.3 ^{0}F post-uprate liner temperature conservatively represents the penetration piping temperature, and the previous Generic Letter 96-06 conclusions remain valid since the piping temperatures post-uprate will be less than those evaluated in the original evaluation.

NRC Question 17

In Reference 3, with regard to the impact of power uprate on the spent fuel pool (SFP) cooling and cleanup system, you stated that the uprate analyses have been performed by revising the single active failure assumption to be a loss of just a single SFP cooling pump. Provide a justification for the deviation from the single active failure assumption described in the FSAR that assumes the loss of one of the two cooling trains for SFP (a pump and heat exchanger). Also, for SFP conditions concurrent with a design basis LOCA, provide the maximum calculated SFP temperature.

CP&L Response

The statement in referenced by the question is in the "Refueling Offload Conditions subsection and is as follows:

> *The Uprate analyses have been performed by revising the single active failure assumption to be a loss of just a single SFP cooling Pump.*

This statement is specific to core offload hold times, which are based on the CCW supply temperature and the CCW and SFP cooling configuration established for the core offload. The Spent Fuel Pool (SFP) cooling system consists of two 100% pumps and two 100% heat exchangers. A 12" cross-connect exists so that a single SFP cooling pump can provide flow through both SFP heat exchangers in parallel (Refer to FSAR Figure 9.1.3-1). The CCW supply to SFP cooling is provided from a single non-essential header and the shell side of the respective SFP heat exchangers can be placed in parallel (refer to FSAR Figure 9.2.2-05). The active component in SFP cooling is the cooling pump. A single failure of a SFP cooling pump does not eliminate the availability of both SFP heat exchangers.

During the design basis LOCA, cooling of the SFP the analysis assumes CCW to SFP cooling would be interrupted for a minimum of 5 hours. At 5 hours, the highest temperature (in SFP A/B) is conservatively estimated to be 144.8 ^{0}F. The time remaining prior to reaching 150°F is 1.2 hours. This time is sufficient to restore SFP cooling. Once CCW flow is restored, the equilibrium SFP temperatures are conservatively calculated to be 140 ^{0}F and 128 ^{0}F for the SFP A/B and SFP C/D, respectively.

REFERENCES

1. Balance of Plant (BOP) Licensing Report, Enclosure 7 to Serial: HNP-00-142, dated October 4, 2000

2. WCAP-15398, Westinghouse Proprietary Class 2C, NSSS licensing report, Enclosure 8 to Serial: HNP-00-142, dated October 4, 2000

3. NSSS Licensing Report, Enclosure 6 to Serial: HNP-00-142, dated October 4, 2000.