PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

10 CFR 50.90

LR-N07-0069
LCR H05-01, Rev. 1
March 30, 2007

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

Hope Creek Generating Station
Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: Response to Request for Additional Information
Request for License Amendment - Extended Power Uprate

Reference: 1) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, September 18, 2006
2) Letter from USNRC to William Levis, PSEG Nuclear LLC, March 13, 2007

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

In Reference 2, the NRC requested additional information concerning PSEG's request. Attachment 1 to this letter restates the NRC questions and provides PSEG's response to each question.

PSEG has determined that the information contained in this letter and attachment does not alter the conclusions reached in the 10CFR50.92 no significant hazards analysis previously submitted.

There are no regulatory commitments contained within this letter

A001

Attachment 1 contains information proprietary to General Electric Company (GE). GE requests that the proprietary information in Attachment 1 be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4) and 2.390(a)(4). An affidavit supporting this request is included with Attachment 1.

A non-proprietary version of the document is provided in Attachment 2.

Should you have any questions regarding this submittal, please contact Mr. Paul Duke at 856-339-1466.

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

Executed on 3/30/07
(date)

Sincerely,

George P. Barnes

George P. Barnes
Site Vice President
Hope Creek Generating Station

Attachments (3)
1. Response to Request for Additional Information (proprietary)
2. Response to Request for Additional Information (non-proprietary)
3. Plant Layout Drawings - RAI 11.7

cc: S. Collins, Regional Administrator – NRC Region I
J. Shea, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
K. Tosch, Manager IV, NJBNE

PROPRIETARY INFORMATION NOTICE

This enclosure contains proprietary information of the General Electric Company (GE) and is furnished in confidence solely for the purpose(s) stated in the transmittal letter. No other use, direct or indirect, of the document or the information it contains is authorized. Furnishing this enclosure does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GE disclosed herein or any right to publish or make copies of the enclosure without prior written permission of GE. The header of each page in this enclosure carries the notation "GE Proprietary Information."

The GE proprietary information is identified by [[double underlines inside double square brackets{3}]]. The superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

General Electric Company

AFFIDAVIT

I, Robert E Brown, state as follows:

(1) I am General Manager, Regulatory Affairs, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of the GE-HCGS-EPU-668, Edward D. Schrull (GE) to Larry Curran (PSEG), *Transmittal - Proprietary Content of Hope Creek Letter, LCR H05-01, Rev. 1 - RAIs 11.1 and 3.5d*, GE Proprietary Information, dated March 30, 2007. The Enclosure 1 (*Transmittal - Proprietary Content of Hope Creek Letter, LCR H05-01, Rev. 1 - RAIs 11.1 and 3.5d*) proprietary information is delineated by a double underline inside double square brackets. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

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

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

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

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed information about the results of analytical models, methods and processes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of coolant activation products in the GE Boiling Water Reactor ("BWR"). The development and approval of the BWR coolant activation products analysis processes was achieved at a significant cost to GE, on the order of several thousands of dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

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

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

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

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 30th day of March 2007

R. E. Brown

Robert E. Brown
General Electric Company

Hope Creek Generating Station
Facility Operating License NPF-57
Docket No. 50-354

Extended Power Uprate

Response to Request for Additional Information

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

In Reference 2, the NRC requested additional information concerning PSEG's request. Each NRC question is restated below followed by PSEG's response.

11) Health Physics Branch (IHPB)

11.1 In the Hope Creek Power Uprate Safety Analysis Report (PUSAR), Section 8.4.1 (page 8-5) you state that Nitrogen Isotope (N^{16}) in the turbine components is expected to increase approximately 16% for a 20% increase in steam flow. On the basis of previous EPU calculations, the activity of N^{16} in steam leaving the reactor pressure vessel is expected to increase in proportion to the power level increase (15% for Hope Creek). Since the steam flow also increases in proportion to the power level increase (i.e., by 15%), the transit time for the N^{16} to reach the major source components that contribute significantly to the skyshine dose in the turbine building is reduced. Therefore, the N^{16} activity in the steam in those areas of the turbine building which contribute to skyshine should increase both in proportion to the power level increase and due to the reduced decay time between the reactor pressure vessel and the turbine building components.

a) Justify your reasoning for stating that the N^{16} activity in turbine components will increase by 16% instead of a higher percentage.

Response
The rates of production of coolant activation products are proportional to power. As a result, the activation products, observed in the reactor water, increase in approximate proportion to the increase in thermal power. However, as stated in PUSAR Section 8.4.1, [[

]].

Nevertheless, the radiation field resulting from activation products will increase with EPU primarily due to the increased steam flow and the resultant decrease in transit time for the activation products from the reactor pressure vessel (RPV) nozzle to the turbine complex. Since these activation products typically have extremely short half-lives, on the order of seconds, the decrease in transit time will result in a measurable increase in downstream activity.

The increase in N^{16} source strength in various steam components mainly depends on the reduction of the transit time to the steam components, which in turn depends on the length of piping between the RPV nozzle and the steam components. The post-EPU plant-specific N^{16} transit times for various steam components were calculated for operation at 3952 MWt. The resulting radiation exposure due to the increased N^{16} source strength in the various steam components is expected to increase 16%. Additional analysis determined that in-plant radiation exposures would be expected to increase by 12% for operation at 3840 MWt (115% CLTP).

b) Justify your reasoning for stating that the steam flow will increase by 20% for a 15% power uprate.

Response
Steam flow will increase approximately 15% for a 15% power uprate. However, the analyses of N^{16} related activity and dose increases described in PUSAR section 8.4.1 were performed for a core thermal power level of 3,952 MWt, representing a 20% increase from the original licensed thermal power level of 3,293 MWt

11.2 In the Hope Creek PUSAR, Section 8.4.1 (page 8-5) you state that the N^{16} levels are expected to increase in the turbine components due to EPU. Verify that the expected increase in dose rates from N^{16} does not create new radiation, or high radiation areas around condensate systems and components in the turbine building.

Response
The 16% increase in the N^{16} related radiation exposure is applied to all steam and condensate bearing systems/component rooms located in the Turbine Building complex. The resulting dose rates in the affected rooms are expected to remain within the applicable radiation zone allowable dose rate limits.

11.3 In the Hope Creek PUSAR, Section 8.4.2 (page 8-6) you state that, although activated corrosion products and fission products are expected to increase as a result of EPU, their post-EPU concentrations will not exceed the design basis concentrations. Provide the expected percentage increases in the concentrations of activated corrosion products and fission products in both the steam and in the water and compare this with the design basis concentration levels.

Response
The calculation of post-EPU reactor coolant water and steam isotopic activity concentrations used ANSI/ANS-18.1-1999 methodology for an assumed core thermal power level of 3,952 MWt. The comparison with the corresponding design basis concentrations is shown in Tables 11.3-1 and 11.3-2.

Table 11.3-1
Noble Gas Activity Release Rate

Isotope	**Steam Release Rate**	
	Design Basis µCi/s	**ANS 18.1/99 Estimate µ Ci/s**
Class 1 - Noble Gases		
Kr-83m	2.9E+03	1.1E+03
Kr-85m	5.6E+03	2.1E+03
Kr-85	2.0E+01	9.0E+00
Kr-87	1.5E+04	5.6E+03
Kr-88	1.8E+04	6.5E+03
Kr-89	1.8E+02	6.5E+01
Xe-131m	1.5E+01	7.4E+00
Xe-133m	2.8E+02	1.1E+02
Xe-133	8.2E+03	3.1E+03
Xe-135m	6.9E+03	2.5E+03
Xe-135	2.2E+04	8.2E+03
Xe-137	6.7E+02	2.6E+02
Xe-138	2.1E+04	7.8E+03
Total	**97,765**	**37,423**

Table 11.3-2
Coolant & Steam Isotopic Activity Concentration

Isotope	Coolant Concentration		Steam Concentration	
	Design Basis μCi/g	ANS 18.1/99 Estimate μCi/g	Design Basis μCi/g	ANS 18.1/99 Estimate μCi/g
Class 2 – Halogens				
I-131	1.3E-02	2.0E-03	2.6E-04	4.5E-05
I-132	1.2E-01	2.0E-02	2.4E-03	4.3E-04
I-133	8.9E-02	1.4E-02	1.8E-03	3.0E-04
I-134	2.4E-01	3.7E-02	4.8E-03	8.1E-04
I-135	1.3E-01	2.0E-02	2.6E-03	4.4E-04
Br-83	1.5E-02			
Br-84	2.7E-02			
Br-85	1.7E-02			
Class 3 - Cesium, Rubidium				
Rb-89		3.9E-03		1.2E-05
Cs-134	1.6E-04	2.3E-05	1.6E-07	6.8E-08
Cs-136	1.1E-04	1.5E-05	1.1E-07	4.6E-08
Cs-137	2.4E-04	6.1E-05	2.4E-07	1.8E-07
Cs-138	1.9E-01	7.9E-03	1.9E-04	2.4E-05
Ba-137m		6.1E-05		1.8E-07
Class 4 - Water Activation Products				
N-13	4.0E-02			
N-16	4.0E+01	4.8E+01(1)	5.0E+01	2.5E+02(2)
N-17	6.3E-03			
O-19	6.9E-01			
F-18	4.0E-03			
Class 5 – Tritium				
H-3		1.0E-02		1.0E-02
Class 6 - Other Nuclides				
Na-24	2.0E-03	1.7E-03	2.0E-06	5.0E-06
P-32	2.0E-05	3.4E-05	2.0E-08	1.0E-07
Cr-51	5.0E-04	2.5E-03	5.0E-07	7.6E-06
Mn-54	4.0E-05	3.0E-05	4.0E-08	8.9E-08
Mn-56	5.0E-02	2.1E-02	5.0E-05	6.2E-05
Fe-55		8.5E-04		2.5E-06
Fe-59	8.0E-05	2.5E-05	8.0E-08	7.6E-08
Co-58	5.0E-03	8.5E-05	5.0E-06	2.5E-07
Co-60	5.0E-04	1.7E-04	5.0E-07	5.1E-07
Ni-63		8.5E-07		2.5E-09
Cu-64		2.5E-03		7.5E-06
Ni-65	3.0E-04		3.0E-07	
Zn-65	2.0E-06	8.5E-04	2.0E-09	2.5E-06

Table 11.3-2
Coolant & Steam Isotopic Activity Concentration

Isotope	Coolant Concentration		Steam Concentration	
	Design Basis μCi/g	ANS 18.1/99 Estimate μCi/g	Design Basis μCi/g	ANS 18.1/99 Estimate μCi/g
Zr-97	3.2E-05		3.2E-08	
Zn-69m	3.0E-05		3.0E-08	
Sr-89	3.1E-03	8.5E-05	3.1E-06	2.5E-07
Sr-90	2.3E-04	5.9E-06	2.3E-07	1.8E-08
Y-90		5.9E-06		1.8E-08
Sr-91	6.9E-02	3.3E-03	6.9E-05	1.0E-05
Sr-92	1.1E-01	8.2E-03	1.1E-04	2.5E-05
Y-91		3.4E-05		1.0E-07
Y-92		4.9E-03		1.5E-05
Y-93		3.3E-03		1.0E-05
Zr-95	4.0E-05	6.8E-06	4.0E-08	2.0E-08
Nb-95	4.2E-05	6.8E-06	4.2E-08	2.0E-08
Mo-99	2.2E-02	1.7E-03	2.2E-05	5.1E-06
Tc-99m	2.8E-01	1.7E-03	2.8E-04	5.1E-06
Tc-101	1.4E-01		1.4E-04	
Ru-103	1.9E-05	1.7E-05	1.9E-08	5.1E-08
Rh-103m		1.7E-05		5.1E-08
Ru-106	2.6E-06	2.5E-06	2.6E-09	7.6E-09
Rh-106		2.5E-06		7.6E-09
Ag-110m	6.0E-05	8.5E-07	6.0E-08	2.5E-09
Te-129m	4.0E-05	3.4E-05	4.0E-08	1.0E-07
Te-131m		8.4E-05		2.5E-07
Te-132	4.9E-02	8.4E-06	4.9E-05	2.5E-08
Ba-139	1.6E-01		1.6E-04	
Ba-140	9.0E-03	3.4E-04	9.0E-06	1.0E-06
La-140		3.4E-04		1.0E-06
Ba-141	1.7E-01		1.7E-04	
Ce-141	3.9E-05	2.5E-05	3.9E-08	7.6E-08
Ba-142	1.7E-01		1.7E-04	
Ce-143	3.5E-05		3.5E-08	
Pr-143	3.8E-05		3.8E-08	
Ce-144	3.5E-05	2.5E-06	3.5E-08	7.6E-09
Pr-144		2.5E-06		7.6E-09
Nd-147	1.4E-05		1.4E-08	
W-187	3.0E-03	2.5E-04	3.0E-06	7.6E-07
Np-239	2.4E-01	6.7E-03	2.4E-04	2.0E-05

Table 11.3-2 Notes:

1. The post-EPU reactor coolant N-16 concentration is expected to increase, which impacts the normal radiation exposure in the reactor building, specifically for the RWCU components. The existing 40-year normal integrated doses in the RWCU equipment rooms remain bounding due to the substantial radioactive decay associated with the protracted N-16 transit times to various RWCU components such that the resulting increase in the post-EPU N-16 related doses become negligible.

2. The post-EPU main steam N-16 concentration includes the 5 times increase due to the Hydrogen Water Chemistry (HWC). The post-EPU N-16 related radiation exposures in the turbine building (TB) complex is calculated in H-1-ZZ-MDC-1930 using the TB radiation exposure data measured during the full scope implementation of the HWC. Therefore, the HWC related N-16 concentration increase is included in the post-EPU exposure assessment.

11.4 In the Hope Creek PUSAR, Section 8.5 (page 8-7) you state that the post-EPU occupational radiation levels in most of the affected plant areas are expected to increase by less than 20%.

a) Justify your statement that radiation levels would increase by approximately 20% when the proposed power uprate is for 15%.

<u>Response</u>
The PUSAR reports the dose increases that were calculated for a core thermal power level of 3,952 MWt, representing a 20% increase to the original licensed thermal power level of 3,293 MWt. These dose increases are greater than that expected for the proposed rated thermal power level of 3,840 MWt (115% CLTP).

b) List any plant areas where you would expect the dose rates to increase by greater than the percentage of the proposed power uprate.

<u>Response</u>
The radiation exposure in all affected plant areas is not expected to increase greater than the proposed 15% EPU. The N-16 radiation exposure increase in the various steam components for operation at 3,840 MWt is not expected to increase more than 12%.

c) Describe what measures you plan to take (e.g., changes to permanent and or temporary shielding, changes to access controls, change to work packages) in areas where dose rates are expected to increase following EPU to maintain worker doses ALARA and within the occupational dose limits of 10 CFR Part 20.

<u>Response</u>
The post-EPU increase in the radiation exposure in various areas housing the steam components remains within the radiation zone allowable dose

rate limits. Therefore, no additional measures are required to maintain the plant exposure ALARA. The post-EPU plant operation and maintenance activities will be controlled by the existing radiation protection and ALARA procedures.

11.5. Describe what impact you expect the proposed power uprate will have on the annual collective doses at Hope Creek and provide an estimate of the occupational dose that will result from the plant modifications that will be needed to support the implementation of the proposed power uprate.

Response

The majority of the plant modifications for EPU have been completed. The remaining modifications, including HP turbine replacement and installation of additional strain gages and vibration monitoring instrumentation, are planned to be completed during the refueling outage preceding EPU implementation. These plant modifications will be made when the N-16 activity in the steam lines has decayed for more than a few hours after plant shutdown, which will completely eliminate any additional N-16 related radiation exposure. Approximately 4.1 person-rem were received for the low-pressure turbine replacement, completed in RF12 (Fall 2004). Approximately 4.2 person-rem were received for strain gage installation completed in RF13 (Spring 2006).

The person-rem exposure for installing the remaining EPU related plant modifications is not expected to exceed the exposure received for the modifications already completed, because the remaining scope is small in comparison to the completed modifications. Since the remaining modifications will be installed prior to EPU implementation, they are not subject to the post-EPU N-16 related dose increase.

The post-EPU annual Person-Rem dose from normal plant operation is expected to increase by no more than 16% and the post-EPU gamma dose rate from the spent fuel assembly is expected to increase by 15%, corresponding to the proportional 15% increase in the power level. The HCGS licensing basis estimate of total annual in-plant exposure from direct radiation is 914 person-rem (UFSAR Section 12.4.1.3.8 and Table 12.4-5). The effective ALARA practice at HCGS has reduced the annual person-rem exposure well below this licensing basis estimate, as evidenced by the historical annual exposures documented in Table 11.5-1. The HCGS ALARA program will continue to ensure that individual exposures are maintained within acceptable limits.

Table 11.5-1
Historical Annual Person-Rem – Hope Creek

Year	Refueling Outage Dose	Days	Forced Outage Dose	Days	Normal Operations Dose	Days	Total Annual Person-Rem Dose
2002	0.00	0.00	0.62	10.00	21.87	355.00	22.49
2003	87.02	30.00	14.54	44.00	34.17	291.00	135.73
2004	173.82	68.00	14.55	37.00	40.68	260.00	229.05
2005	10.95	26.00	20.41	33.00	35.62	306.00	66.97
2006	92.96	31.00	0.00	0.00	44.43	334.00	137.39

11.6. In the Hope Creek PUSAR, Section 8.5 (page 8-6) you state that a post-EPU radiation assessment in the turbine building complex to evaluate the effects of the proposed EPU on area dose rates was completed.

a) Discuss how you plan to verify post-EPU dose rates throughout the plant. Will you be conducting radiation surveys of selected plant areas as part of the EPU startup and test plan?

Response
Plant Area Radiation and Process Monitors will be monitored at 90% and 100% of current licensed thermal power (CLTP) and at 2.5% reactor power intervals above CLTP. As part of the power ascension test plan, normally accessible areas adjacent to steam affected areas in the Turbine and Reactor Buildings and the Radwaste area of the Auxiliary Building will be surveyed at specified intervals of reactor power. Postings will be verified during these surveys. As required to support power ascension test program activities, escorted entries will be made to areas near the main turbine and associated support systems.

b) Have you identified plant areas that may require changes in radiation shielding or zone designations?

Response
No areas have been identified that may require changes in radiation shielding or zone designations.

c) Please provide a listing of plant areas where you will conduct radiation surveys following the proposed EPU implementation and describe your criteria for selecting these areas.

Response
Radiation surveys will be performed in normally accessible areas adjacent to steam affected areas in the Turbine and Reactor Buildings and the Radwaste area of the Auxiliary Building. The survey plan will be developed from the plan used for implementation of the 35 scfm hydrogen injection rate for Hydrogen Water Chemistry. This is considered adequate because actual dose rates for EPU implementation are expected to be bounded by dose rates observed for operation with 35 scfm hydrogen injection.

Surveys will be conducted in the following plant areas:

- Turbine Building 171 Elevation
- Turbine Building 137 Elevation
- Turbine Building 120 Elevation
- Turbine Building 102 -111 Elevations
- Turbine Building 77 - 87 Elevations
- Turbine Building 54 Elevation
- Service/Radwaste Area 54 Elevation
- Reactor Building 132 Elevation
- Reactor Building 102 Elevation
- Reactor Building 54 Elevation

11.7. Provide the following clarifying information regarding the information contained in Table 8-1 (page 8-9) of the Hope Creek PUSAR:

a) Table 8-1 has a column which shows allowable occupancy times for each of the vital areas listed. The title "allowable occupancy" can imply that this is the maximum time that a person can be in the area before exceeding the 5 REM, in Title 10 of the Code of Federal Regulation (10 CFR) Part 50, Appendix A, General Design Criteria for Nuclear Power Plants (GDC) 19, dose limit. Verify that the times listed in this column are the estimated times needed to complete the mission in each of the vital areas listed.

Response
The Hope Creek plant specific analysis to comply with the NUREG-0737, Item II.B.2 requirement has been performed in calculation H-1-ZZ-MDC-1927, Rev 0 for the vital area mission doses.

The "allowable occupancy" times listed in PUSAR Table 8-1 are the maximum times permitted in the listed vital areas before exceeding the allowable limit of 5 rem TEDE dose per 10 CFR 50.67, instead of 5 rem whole body dose per 10 CFR 50, Appendix A, GDC 19 because the Hope Creek operating license has been amended for the Alternate Source Term

(AST) on October 3, 2001 via License Amendment No. 134, that replaced the whole body and thyroid dose criteria by TEDE (total effective dose equivalent) criteria.

The estimated mission times to perform each of the vital functions are extracted from the Hope Creek original plant-licensing basis UFSAR Table 12.3-3, Rev 0, and included in the first column of Table 11.7.a-1. Table 11.7.a-1 demonstrates that the mission times are considerably less than the allowable occupancy times for each of the vital functions. However, actual occupancies would be based on radiation surveys, and use of respirators to reduce the inhaled dose would be controlled by plant procedures.

Table 11.7.a-1
Vital Access Mission Time

	Post-LOCA Dose Rate (rem/hr)				
Vital Access Area Locations (and Mission Times)	**Whole Body (A)**	**TEDE w/o FFNPR (B)**	**Mission Dose w/ FFNPR (rem) (C)**	**Allowable Mission W/ FFNPR (hr) (D)**	**Reference Layout Drawing****
Operational Support Center	8.30E-03	1.50E+00	*	*	*
Guardhouse	5.90E-03	3.10E-01	*	*	*
Diesel Generator & Accessories (1 hour)	3.60E-02	1.30E+00	3.60E-02	138.89	P-0053-0, P-0054-0, & P-0055-0
FRVS RMS Skid (1.5 hours)	5.80E-02	5.10E+00	8.70E-02	86.21	P-0055-0, P-0034-0, P-0035-0, & P-0036-0
Remote Shutdown Panel Area (1 hr)	5.80E-02	5.10E+00	5.80E-02	86.21	P-0055-0 & P-0035-0
HP/Access Control Point (1 hour)	5.80E-02	5.10E+00	5.80E-02	86.21	P-0055-0 & P-0035-0

* Requiring Frequent Access. Dose Rate Provided for Information Only
$C_i = A_i \times T_i$
$D_i = 5/A_i$
** Reference Layout Drawings Show Access To/From Control Room To Vital Function Locations

b) Verify that the whole body and total effective-dose equivalent (TEDE) dose rates shown in table 8-1 are the maximum post-accident dose rates reached in each of the vital areas listed and may not [necessarily] be the dose rates in the vital area when the area needs to be accessed to perform post-accident vital area functions.

Response
The post-LOCA whole body and TEDE dose rates shown in PUSAR Table 8-1 are the maximum dose rates in each of the vital access areas occurring at any time during the accident. These maximum dose rates are not dependent on the times for performing the vital functions. The maximum dose rates occur at different times during the initial phase of accident, depending on the location of the vital areas relative to the post-accident release points, and the relative locations of air intakes for the buildings housing the vital equipment.

c) For each of the vital areas listed in Table 8-1, provide a brief description of why the area is classified as a vital area (i.e., what is the vital area function which needs to be performed in the area) and provide plant layout maps indicating the location of each of the vital areas listed in Table 8-1

Response
Vital areas are defined in NUREG-0737, Item II.B.2, as those "which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident." The expected vital functions to be performed at various vital access location are listed in Table 11.7.c-1:

Table 11.7.c-1
Vital Access Area Locations and Functions

Vital Access Area Locations	Vital Function To Be Performed
Operational Support Center	Emergency Assembly Point
Guardhouse	Control Security Access for Accountability
Diesel Generator & Accessories	Control Operation of Diesel Generator During Emergency Operation
FRVS RMS Skid	Post-Accident Sampling To Determine Core Degradation
Remote Shutdown Panel Area	Alternate Location for Plant Safe Shutdown
HP/Access Control Point	Plant Control Access Point
Controlled Hot Chemical Lab	Lab To Analyze Post-accident Samples

The Post Accident Sampling System (PASS) has been removed from the Hope Creek licensing basis as a result of Hope Creek license amendment 149, dated January 29, 2004. The NRC approved the elimination of PASS based on the contingencies plans described in the Hope Creek UFSAR Section 9.3.2.1.2. Therefore, the PASS locations are omitted from Tables 11.7.a-1 and 11.7.c-1.

HCGS UFSAR Table 12.3-3 identifies only four (4) vital access areas requiring infrequent access:

- Diesel Generator & Accessories,
- FRVS RMS Post-Accident Sample Skid,
- Controlled Hot Chemical Lab, and
- FRVS Sample Transport Path

Plant layout drawings showing the locations of these vital areas are provided with this RAI response. The locations of each of the vital access areas are considered in the analysis in order to establish the atmospheric dispersion factors (χ/Qs) for the air intake locations. These dispersion factors are used to calculate the bounding submergence and inhaled dose rates listed in PUSAR Table 8-1.

d) Verify that the vital area missions for each of the vital areas listed in Table 8-1 can be accomplished following an accident post-CPPU without exceeding the 5 person-REM criteria specified in 10 CFR Part 50, Appendix A, GDC 19. In calculating the mission doses for each of the vital areas, the doses received to access and exit from the area should be included in the total vital area mission dose estimate.

Response

The vital area missions for each of the vital areas listed in PUSAR Table 8-1 can be accomplished following an accident post-CPPU without exceeding the 5 person-REM criteria specified in 10 CFR Part 50, Appendix A, GDC 19.

Compliance to NUREG-0737, Section II.B.2, assures the shielding adequacy necessary to reduce the whole body dose (i.e., external dose) to an operator to perform the vital function in a given mission time to less than the allowable limit of 5 rem whole body dose. The Hope Creek original plant-licensing basis for the vital access area mission times for various vital functions are listed in UFSAR Table 12.3-3, Rev 0. These times are shown in column 1 of Table 11.7.a-1.

As shown in Table 11.7.a-1, all vital area functions can be performed with mission doses that are less than a small fraction of the allowable regulatory limit of 5 rem whole body dose per 10 CFR 50, Appendix A, GDC 19 and NUREG-0737, Item II.B.2. As shown in Table 11.7.a-1, the resulting allowable occupancy time before exceed the 5 rem dose limit is far greater than the mission time proposed in the original licensing basis.

11.8. Section 6.3.3 (pages 6-4,5) of your submittal states that the post-EPU radiation exposures in accessible areas adjacent to the sides or bottom of the spent fuel pool (SFP) are expected to be within the allowable dose rate limit of the existing radiation zone designation. Discuss any plans that you may have (such as shuffling of spent fuel assemblies in the SFP so that the older assemblies are located at the perimeter of the SFP) to minimize the effects of the storage of the higher irradiated spent fuel assemblies in the SFP on dose rates in areas surrounding the SFP.

Response
The normal radiation levels around the SFP are expected to increase primarily during fuel handling operations during the refueling outage. The plant-specific post-EPU radiation exposures around the SFP are analyzed in calculation H-1-ZZ-MDC-2059, Rev 1, using the spent fuel assembly activity with a discharged bundle exposure (fuel burnup) of 58 GWD/MTU. The resulting dose rates at various locations (including 3 feet below the SFP floor, at the sidewall, and 8 feet above the water surface on the bridge) are all well within the radiation zone allowable dose rate limits. Since the analysis is performed for a freshly discharged spent fuel assembly (SFA) and with minimum spent fuel water shielding based on the limitation of movement of the refueling bridge, the resulting exposures are the highest dose rates expected during the refueling. The old SFAs discharged from the previous outages and stored on the spent fuel racks below 23' water level (Tech Spec LCO 3.9.9) contribute small dose rates to the surrounding areas due to the associated concrete shielding (calculation H-1-ZZ-MDC-2059, Rev 1, Table 15). Therefore, there is no need for any plan to rotate the spent fuel assemblies to further reduce the existing small radiation exposures from the stored SFAs.

11.9. Discuss what affects the proposed EPU will have on the whole body dose to the public with respect to the 25 mrem per year dose limits of 40 CFR 190.

Response
The post-EPU annual offsite dose is 9.3 mrem, which is less than the allowable annual dose limit of 25 mrem in 40 CFR 190.

12) Component and Performance and Testing Branch (CPTB)

12.1. PSEG is requested to discuss the plans to implement the Inservice Testing (IST) Program for Hope Creek that incorporates appropriate changes in light of applicable EPU operating conditions. In particular, the licensee is requested to discuss with three examples of each type, if applicable, the evaluation of the impact of EPU conditions on the performance of safety-related pumps, power-operated valves, check valves, safety or relief valves, including consideration of changes in ambient conditions and power supplies (as applicable), and dynamic restraints; and to indicate any resulting component or support modifications, or adjustments to the IST Program, resulting from that evaluation.

Response

Summary of IST Program Impact

Since the EPU maintains the same reactor dome pressure, the primary impacts are higher main steam and feedwater flows, which also raise the flow rates in other power block systems such as the condensate and extraction steam systems. EPU will result in the primary condensate, secondary condensate, and reactor feed pump operating at higher flows. There are no increases in the required flow rates for any safety related systems.

For the postulated design basis LOCA, peak drywell pressure (Pa) increases from 48.1 to 50.6 psig. This increase in Pa affects the test pressure for local leakage testing of containment isolation valves in the IST program, and is also a consideration for defining maximum allowable stroke time of inside-containment Main Steam Isolation Valves (MSIVs), as described below in the "Power-Operated Valve" section.

PUSAR Sections 6.1 and 6.2 respectively address AC and DC power supplies. Safety-related electrical loads, including diesel generator loads, are not changed. There are no IST program changes associated with EPU power supply changes.

EPU impact on mechanical equipment with respect to ambient conditions is addressed in response to RAIs 12.5 and 12.6, and did not result in any IST program impact.

With respect to dynamic restraints, snubber qualification considerations due to EPU environmental conditions are also addressed in response to RAI 12.5, that concludes no changes to snubber maintenance or surveillance are required. In order to accommodate the increased Turbine Stop Valve (TSV) closure transient loads due to higher Main Steam (MS) flows, six Main Steam pipe supports were modified to ensure the current UFSAR design allowables are met for EPU conditions. These support modifications did not modify, add or delete any snubbers, nor did they impact the snubber functional test program defined in Technical Specification 3/4.7.5. If final reconciliation of any outstanding piping or support open items (e.g., PUSAR Section 3.5 and 10.1.2; potential support modifications noted in Attachment 5 to our September 18, 2006 request) affects snubbers, then potential changes to snubber inspection or testing would be addressed as part of the HCGS Engineering Change Process.

Examples of Safety-Related Pumps

Examples of safety-related pumps at HCGS are

High Pressure Coolant Injection (HPCI)
Low Pressure Coolant Injection (LPCI)
Core Spray (CS)

Based on the PUSAR Section 4.2 evaluations of system performance, no changes to IST of these pumps result from EPU.

Examples of Power-Operated Valves

Inboard MSIVs (HV-F022A, HV-F022B, HV-F022C, HV-F022D) – Closure of these valves in response to a containment isolation signal is achieved via spring force and a compressed gas piston-cylinder mechanism. Compressed gas under the piston is vented to the drywell atmosphere during the closure stroke. Therefore, the effect of drywell pressure under accident conditions (Pa) is considered in determining maximum allowable stroke time. The IST procedure defining the inboard MSIVs maximum stroke time is being revised to account for the increase in Pa. MSIV leak rate testing is also revised to reflect the increased Pa. As summarized in response to RAI 12.4, EPU conditions do not adversely affect the performance of these valves.

For other power-operated valves in the IST program, the only change in inservice testing is to increase the local leak rate test conditions due to the increase in Pa. Examples of evaluation of safety-related power-operated valves under EPU conditions are provided for the EDG lube oil cooler SACS valves and the Drywell Purge Exhaust containment isolation valves, in response to RAI 12.4.

Examples of Check Valves

FW isolation check valves (AE-V007 / AE-V003) are required to prevent reverse flow from the reactor vessel to outside containment. These check valves are safety-related inboard containment isolation valves and are leak rate tested at Pa. These valves are acceptable for operation under EPU conditions including increased FW flow. IST of these valves is affected by the increased Pa that defines local leak rate test conditions.

Standby Liquid Control System (SLCS) header check valve BH-V029 is an inside-containment isolation valve. As noted in Section 6.5 of the PUSAR, the

SLCS performance is not affected by EPU. IST of this valve is affected by the increase in Pa that defines local leak rate test conditions.

Examples of check valves with resilient seats that were evaluated and determined to require no change in maintenance or surveillance due to EPU conditions, are provided in response to RAI 12.5.

Examples of Safety or Relief Valves

HCGS Safety Relief Valves (SRVs) provide nuclear system overpressure protection and Automatic Depressurization System (ADS) capability as an Emergency Core Cooling System (ECCS) function. As described in PUSAR sections 3.1 and 4.2.5 respectively, the EPU does not impact the SRVs' setpoints or tolerances for overpressure protection or ADS.

Pressure vessel re-rating and resultant relief valve setpoint increases were implemented for the Moisture Separators and #5 feedwater heaters to support operation at EPU conditions. These relief valves are non-safety related and therefore do not affect the IST program.

12.2. In the Hope Creek PUSAR, Section 8.5 (page 8-6) you state that process parameters of temperature, pressure, and flow for motor-operated valves (MOVs) within the scope of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," were reviewed; and minor changes were identified as a result of EPU conditions. You also state that MOV calculations will be revised as necessary. PSEG is requested to discuss with examples its evaluation of safety-related MOVs within the programs established in response to GL 89-10 and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," at Hope Creek for the potential impact from EPU operation, including the impact of increased process flows on operating requirements and increased ambient temperature on motor output.

Response

Effects of increased process flow:

Flow rates for EPU operation increase for four systems with MOVs in the GL 89-10 program: Main Steam (MS), Reactor Recirculation (RR), Feedwater (FW) and Safety and Turbine Auxiliary Cooling (STACS).

For the MS and RR systems, fluid momentum effects were not considered in the DP calculations (gate & globe valves), therefore there is no impact.

For the two MOVs in the FW system, the fluid momentum effects were not considered because the maximum DP is zero, therefore there is no impact.

For the STACS system butterfly valve MOVs, the fluid momentum impact was considered to be negligible (less than 1%), therefore there is no impact.

Therefore, there is no significant impact on the MOVs in the GL 89-10 and GL 96-05 program due to the increased flow rates.

Effects of increased ambient temperature on motor output:
The increased temperature of the drywell and torus are bounded by existing analyses which use design temperatures. The capability calculations for MOVs in the drywell and torus use temperatures greater than 300 degrees F. The post EPU maximum drywell temperature increased from 291 to 298 degrees. The peak short-term torus temperature increased from 125 degrees to 126 degrees. Therefore, the increase in temperature is considered to have a negligible impact on the MOV motor torque output capability.

12.3. In the Hope Creek PUSAR, Section 4.1.4 you state that the effect of the EPU on the potential for pressure locking and thermal binding under GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," was reviewed. The licensee is requested to discuss with examples its evaluation of safety-related power-operated gate valves in light of any changes in ambient temperature on the potential for pressure locking or thermal binding resulting from EPU operation.

Response
Pressure Locking:

The evaluation of MOV gate valves performed in 1995 identified thirteen MOVs as requiring further evaluation for susceptibility to pressure locking. Ten of these MOVs were subsequently modified by drilling a hole in the disk to relieve bonnet pressure and are no longer susceptible to pressure locking. As discussed in Reference 12.3-1, further evaluation determined pressure locking not to be a concern for the remaining three MOVs, BCHV-F004A and B and BCHV-F009.

Thermal Binding:

The evaluation of MOV gate valves performed in 1995 identified seven valves as requiring further evaluation for susceptibility to thermal binding. For six of the valves, further evaluation determined that the safety function of the valve was not impacted by the thermal binding scenario.

The seventh valve is the FDHV-F001 HPCI steam admission valve. This valve has been determined to not be a concern based on a demonstrated capability to open after a 160 degree drop in temperature. Also the HPCI system was determined to not be impacted by the EPU.

Reference
12.3-1 PSEG letter LR-N96034, Response to Generic Letter 95-07, February 13, 1996

12.4. In the Hope Creek PUSAR, Section 4.1.4 you state the process parameters of temperature, pressure, and flow for air-operated valves (AOVs) were reviewed, and no changes to the functional requirements of any AOVs were identified. PSEG is requested to summarize the approach used and provide three examples of the methodology for evaluation of safety-related AOVs (and solenoid-operated valves, as applicable) for potential impact from EPU operation.

Response
The EPU project analyzed the impact of the extended power uprate on Hope Creek safety related Air Operated Valves (AOVs). Methods used include Design Bases Reviews (DBR) relative to design bases pressure sources and the effect of EPU on system conditions. For AOVs which DBRs have not been completed, the Hope Creek IST Manual, Appendix A, IST Program Submittal documents were reviewed and compared to the effect of the EPU for these AOVs based on the systems condition changes. The EPU effect on DP and flow rate through the valve was evaluated for butterfly valves. Three examples are listed below.

Example 1, Main Steam Isolation Valves (4 inboard/ 4 outboard) were analyzed for the effect of EPU. This analysis concluded that MSIV valve closing time remains within limits and EPU has a negligible impact on the MSIV thermal fatigue usage factor. The evaluation covers both the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements, and the potential effects of EPU-related changes to the safety functions of the MSIVs. The generic evaluation is based on a 20% thermal power increase, (an increased operating dome pressure to 1095 psia, a reactor temperature increase to 556°F, and (4) steam and feedwater flow increases of about 24%). Increased EPU flow rate assists MSIV closure, which results in a slightly faster MSIV closure time. Therefore, the MSIVs are acceptable for EPU operation including isolation performance and valve pressure drop.

Example 2, safety related valves 1EGV-231/1EGHV-2395A, 1EGV-235/1EGHV-2395B, 1EGV-233/1EGHV-2395C and 1EGV-237/1EGHV-2395D, EDG Lube Oil Cooler SACS Valves. These valves open to permit cooling flow to diesel generators A-D. The design basis functions and pressures were reviewed for EPU effect and it was concluded that SACS system pressure is unchanged and there is no SACS flow increase through these valves; therefore, they are not affected by EPU.

Example 3, safety related AOVs 1GSV-024/1GSHV-4952 and 1GSV-026/1GSHV-4950 Drywell Purge Exhaust Containment Isolation valves were reviewed for peak drywell pressure. The capabilities of the isolation actuation devices to perform during normal operations and under post-accident conditions have been determined to be acceptable and are not adversely affected by the EPU. Peak containment pressure will not exceed the valve's maximum design pressure of 65 psig, which conforms to the Design Specification Valve Data Sheet.

12.5. In the Hope Creek PUSAR, Section 10.3, Environmental Qualification, you indicate that safety-related components are to be qualified for the environment in which they are required to operate. In Section 10.3.2, Mechanical Equipment with Non-Metallic Components, you state that the reevaluation of safety-related mechanical equipment with non-metallic components identified some equipment potentially affected by EPU conditions that were resolved by reanalysis. PSEG is requested to provide examples of the range of the non-metallic components in safety-related mechanical equipment effected by the EPU. These examples should include a discussion on the following topics:

a) applicable environmental conditions;
b) required operating life;
c) capabilities of the non-metallic components;
d) basis for the environmental qualification of mechanical equipment; and the
e) surveillance and maintenance program to be developed to ensure functionality during their design life.

Response

The HCGS Mechanical Equipment Qualification (MEQ) program established the capability of active safety-related mechanical equipment to perform its required safety function for the life of the plant including postulated accident conditions. Safety related active mechanical equipment, i.e., equipment which must move or change position to perform its design safety function during a Design Basis Accident (DBA), was included in the program. Pumps, fans and check valves are examples of active safety related mechanical components. Nonmetallic parts used in mechanical equipment include gaskets, diaphragms, seals, lubricating oil or grease, fluids for hydraulic systems, flexible hoses and packing. These age sensitive components were analyzed to ensure that the material can perform their intended function during the postulated normal and accident conditions (e.g., temperature, radiation). MEQ program equipment was determined to either have a 40 year qualified life or be subject to maintenance tasks under current HCGS programs (e.g., Air-Operated Valve program, Motor-Operated Valve program) to ensure the equipment will operate satisfactorily under normal or accident conditions.

Review of the postulated DBA conditions due to EPU reveals that the current temperature and pressure profiles used for evaluation are higher than the temperature and pressure postulated during DBA conditions due to EPU, as tabulated in response to RAI 12.6. The radiation conditions are not bounded. All equipment belonging to MEQ program were evaluated to ensure that the non-metallic parts are capable of performing their intended function during normal and accident conditions. Results of the review indicated that the postulated radiation due to EPU is higher than the radiation damage threshold of the non-metallic parts of Resilient Seated Check Valves (RSCV) and Hydraulic Snubbers. Location Specific Radiation Calculations were performed to determine the postulated radiation dose for this equipment.

1) The mechanical components which required reanalysis were Resilient Seated Check Valves (RSCV), i.e., Main Steam Isolation Valve (MSIV) accumulator check valves; and Hydraulic Snubbers .

   The non-metallic components which required reanalysis are as follows:

   Viton Elastomer Seal O-rings in Resilient Seated Check Valves (RSCVs) used for the inboard and outboard MSIV accumulator check valves.

   Viton Seals and AK-350 Silicon Hydraulic Fluid in LISEGA Hydraulic Snubbers.

2). The applicable radiological environmental conditions that exist during 40 years normal operation followed by a LOCA (340°F and 5.99E7 rads of gamma for RSCVs and 340°F & 3.7E7 rads of gamma for Hydraulic Snubbers).

3). The required operating lives are 40 years of normal life plus required Post Accident Operability Period.

   Resilient Seated Check Valve have Post Accident Operability Period of 2 hours and Hydraulic Snubbers have Post Accident Operability Period of 100 days.

4) The safety related function of RSCVs is to provide the unidirectional flow during the mission times discussed in the Response 3 above. The safety related function of Hydraulic Snubber is to transmit the dynamic forces to supporting structure without restricting the thermal expansion of the supported piping or component during the mission times discussed in the Response 3 above.

5) PSEG calculation No. H-1-ZZ-MDC-2030 has performed location specific radiation dose and determined that the radiation dose postulated at the most conservative location of the RSCV with respect to the radiation sources is below the radiation damage threshold.

PSEG Calculations H-1-ZZ-MDC-2031, 2051, 2052, 2057, 2070, 2071, 2072, and 2073 are performed to calculate the location specific radiation dose for the Hydraulic Snubbers used at Hope Creek. Results of these calculations show that the snubbers are qualified for the postulated environment for the remaining design life of the plant.

6) The location-specific analyses performed for the non-metallic components in the RSCVs and hydraulic snubbers indicate that the calculated doses remain below the radiation damage threshold. Therefore, there is no need to revise the existing surveillance and maintenance program to ensure functionality during their design life.

The MSIV accumulator check valves (RSCVs) are tested as part of Technical Specification surveillance testing of the MSIVs. The RSCVs are tested to verify forward flow and closure capability. Hydraulic snubbers are inspected and tested in accordance with the surveillance requirements of Technical Specification 3/4.7.5.

12.6. In the Hope Creek PUSAR, Section 10.3.3, Mechanical Component Design Qualification, you state that mechanical design of equipment and components in certain systems is affected by operation at EPU conditions due to slightly increased temperatures and, in some cases, flow. Also, you state that the revised operating conditions do not significantly affect the cumulative usage fatigue factors of mechanical components that the increased fluid induced loads on safety-related components and supports are insignificant. PSEG is requested to:

a) Discuss the environmental qualification methods and approaches applied to mechanical equipment (including pumps, power-operated valves, safety-related valves, and check valves) and their supports.

<u>Response</u>
The HCGS Mechanical Equipment Qualification (MEQ) Program applied to age-sensitive, nonmetallic parts used in safety-related active mechanical equipment. Radiation and temperature effects are the primary aging stressors that were considered in the MEQ program. Other mechanical analyses, including fatigue analysis, are separate from the MEQ program, and are described in Sections 3, 4, 10.1 and 10.2 of the PUSAR. For example,

- PUSAR Section 3.8 summarizes the evaluation of the effects of EPU changes on the structural capability of the Main Steam Isolation Valve (MSIV) to meet pressure boundary requirements, and the potential effects of CPPU-related changes to the safety functions of the MSIVs.

- PUSAR Section 10.1.2,High Energy Line Break (HELB), identifies the need to reconcile the cumulative fatigue usage at three Feedwater piping locations.
- PUSAR Section 4.1.4 concludes no changes to the functional requirements of any air-operated valve (AOV) were identified as a result of operating at the EPU conditions. Examples of evaluations to determine the effects of EPU parameter changes on AOVs are provided in response to RAI 12.4.

To establish the environmental qualification of the mechanical equipment under EPU conditions, the impact of Design Basis Accident (DBA) and normal operating conditions was evaluated. The temperature and radiation withstand capability of each non-metallic material is established using material data available in industry to ensure that the mechanical equipment can perform their intended function under the postulated environmental conditions during normal and accident conditions during their design life. Each non-metallic material is evaluated to ensure that it has capability to perform its intended function under the postulated environmental condition. The evaluation considered the maximum temperature postulated during DBA condition and Total Integrated Dose (normal dose for 40 years plus accident dose) to ensure the operability of the mechanical equipment.

b) Provide at least three examples of equipment and components that will experience increased temperatures, flows, and loads resulting from EPU conditions to demonstrate that the impact is insignificant.

<u>Response</u>
The comparison of key parameters during accident conditions used to evaluate the post-EPU impact on the equipment in the MEQ program are listed as follows:

Key Parameter	Bounding Current License Thermal Power (CLTP) Value	Calculated Extended Power Uprate (EPU) Value
LOCA Peak Temperature Inside Drywell (°F)	340	298
LOCA Peak Pressure Inside Drywell (psig)	62	50.6
EQ Radiation Dose (Rad)	7.24E+07	8.40E+07
HELB Peak Temperature Outside	230.7	223.8

Key Parameter	Bounding Current License Thermal Power (CLTP) Value	Calculated Extended Power Uprate (EPU) Value
Primary Containment (°F)		
HELB Peak Pressure Outside Primary Containment (psig)	6.4	3.8

The comparison of bounding MEQ program pressure and temperature to calculated EPU pressure and temperature in the above table indicates that the CLTP MEQ values remain bounding for the EPU. The post-EPU radiation dose is expected to be higher than the CLTP but it remains below the radiation damage threshold for all mechanical equipment except for the equipment described in response to RAI 12.5.

Mechanical Equipment Qualification Program used the worst case environmental condition postulated during accident conditions to demonstrate the capability of the mechanical equipment to perform their intended function. EPU did not impact the worst case temperature and pressure postulated during accident condition. The current specified normal plant temperature, pressure and humidity conditions for zones containing EQ equipment are bounding for EPU. Therefore, there was no impact on mechanical qualification of equipment due to EPU-related changes in environmental conditions.

Examples of mechanical equipment and components that will experience increased temperatures, flows and loads resulting from EPU conditions are as follows.

1. PUSAR Table 1-3 shows an increase in FW system temperature and flow due to EPU. FW isolation check valves (AE-V007 / AE-V003) are required to prevent reverse flow from the reactor vessel to outside containment. These valves are acceptable for operation under EPU conditions including increased FW flow. These valves do not contain age-sensitive, nonmetallic parts that would require evaluation of MEQ-related maintenance changes.

2. As shown in PUSAR Table 1-3, steam flow rate increases due to EPU. PUSAR Table 3-10 shows the maximum percent increase in Turbine Stop Valve (TSV) Transient load due to the flow increase. Pipe support modifications implemented for Main Steam system supports outside containment maintain

conformance to current design allowables for piping and supports with consideration of the higher TSV transient loads following EPU.

3. Operation at EPU conditions would exceed the original design conditions of the 5th point FW heaters (5A/5B/5C) due to increases in extraction steam and moisture separator operating conditions. The 5th point FW heaters were re-rated from a working pressure design rating of 225 psig to 260 psig, and their relief valve setpoints were adjusted accordingly, to support EPU operating conditions. These heaters and relief valves are non-safety-related.

c) Describe the surveillance and maintenance program for mechanical equipment to ensure functionality during their design life.

Response
The HCGS Mechanical Equipment Qualification program applied to active, safety-related mechanical equipment located in potentially harsh environmental areas. The program focused on the effects of the environment on age-sensitive materials and parts. Materials that are not age-sensitive or that are sufficiently resistant to the 40-year plant environment plus the anticipated effects of a design basis event, did not require further evaluation. Current HCGS programs e.g., Air-Operated Valve (AOV), Motor-Operated Valve (MOV) programs, use inspection, test or rebuild activities that provide confidence in the components' ability to function in accordance with their specified requirements during their design life.

13) Containment and Ventilation Branch (SCVB)

13.1 In the Hope Creek EPU application request, Section 4 of Attachment 1, Request for Change to Technical Specifications Extended power Uprate addressed Ultimate Heat Sink (UHS) and the design calculation for UHS temperature limits. It was stated that the Emergency Core Cooling System (ECCS) cooler loads in the UHS temperature limit calculation are based on an updated reactor building GOTHIC model analysis. Describe the GOTHIC model, and sources of heat input (e.g. pump motors, piping, electrical). Did the ECCS cooler loads increase, stay the same or decrease? Response to this question, at least in part, can be included in response to question 13.6.

Response
The Filtration, Recirculation, and Ventilation System (FRVS) and Emergency Core Cooling System (ECCS) cooler heat loads in the current UHS temperature limit calculation are based on an updated reactor building GOTHIC model

analysis. In the evaluation submitted in PSEG letter LR-N98274, "Request for Change to Technical Specifications: Ultimate Heat Sink Temperature Limits", June 12, 1998, it was stated that heat loads were based on a reactor building model that was built in a spreadsheet. The GOTHIC model provides a more accurate representation of actual conditions.

The GOTHIC model and analysis are documented in PSEG calculation 11-0066, revision 7. The GOTHIC software is a PC based program that provides a comprehensive analysis of heat transfer across multiple rooms. The Gothic model uses CPPU drywell temperature, suppression pool temperature, and wet well pressure profiles based on a reactor power level of 102% of 3952 MWt, which conservatively bounds CPPU at 3840 MWt.

Reactor building internal and external heat loads are represented in the model including piping and equipment. Walls, structures, equipment, and piping are modeled, including room coolers, the FRVS cooling coils and the RHR and SACS heat exchangers. The heat loads used in the GOTHIC model include the building heat gain from the outside environment, heat loads from lighting, passive (e.g., tanks, heat exchangers) and active equipment (e.g., energized equipment such as pumps, fans, motors, transformers, substations and MCCs) and piping heat loads. The heat gains from equipment in each room are documented in PSEG calculation GR-0001.

The results for current licensed power and CPPU are compared below. As can be seen, the greatest increase in room temperature occurs in room 4502, 4620 (RWCU Filter Demineralizer Holding Pump Room) and is less than 3°F (2.6°F). Generally, room temperature increases are less than 1°F and the impact on ECCS room coolers and equipment performance is insignificant.

The temperatures for the control volumes that represent the RHR, CS, HPCI, and RCIC rooms remain below 125°F for all cases. Temperature of SACS pump rooms remain below 110°F.

GOTHIC Control Volume	Reactor Building Rooms/Areas Included	Maximum Vapor Temperature (F)--CLTP	Maximum Vapor Temperature (F)--CPPU
1	4101	118.4	118.8
2	4104	109.9	110.2
3	4103	109.5	109.8
4	4105	107.5	107.7
5	4106	117.5	118.2
6	4107	112.6	112.8
7	4108	109.6	109.9
8	4109, 4208	112.2	112.3
9	4110	104.7	104.7
10	4111	107.4	107.5
11	4112	110.1	110.4

GOTHIC Control Volume	Reactor Building Rooms/Areas Included	Maximum Vapor Temperature (F)-- CLTP	Maximum Vapor Temperature (F)-- CPPU
12	4113, 4214	111.2	111.3
13	4114	111.3	111.4
14	4115	114.3	114.7
15	4116	108.1	108.3
16	4117	108.1	108.3
17	4118	107.5	107.6
18	41-01 - Stair	108.2	108.4
19	41-02 - Stair	112.3	112.6
20	41-03 - Stair	112.1	112.4
21	41-04 - Stair	110.0	110.2
22	41-04 - Elevator	104.2	104.3
23	Various	128.7	129.7
24	4201	111.5	111.9
25	4202	103.0	103.4
26	4203	113.1	113.7
27	4205, above 4204	111.7	112.1
28	4207, above 4206	123.8	123.9
29	Various	111.6	112.0
30	4210	107.0	107.4
31	4219	108.5	108.9
32	4215	104.6	104.9
33	4216	107.3	107.8
34	4218, above 4217	104.0	104.4
35	4204	92.0	92.0
36	4206	95.0	95.0
37	4212	91.0	91.0
38	4217	92.0	92.0
39	4301, 4310, above 4313	108.5	108.8
40	4303, above 4304	108.4	108.7
41	4304	109.9	110.2
42	4305	108.4	108.7
43	4307, above 4305	100.9	101.0
44	4309, above stair 41-03	109.5	109.6
45	4311	108.7	108.9
46	4313	112.0	112.3
47	Various	101.6	101.8
48	4326, includes 4333	114.8	116.1
49	4330	106.7	107.4
50	4316, 4518	112.0	112.0
51	4318	107.7	108.5
52	43-01 - Stair	117.4	117.9
53	43-01 - Elevator	112.1	112.7
54	4323, above 4324	106.8	107.5
55	4324	112.8	113.5
56	43-02 - Stair	116.6	117.2
57	4334, above 4311	102.4	102.7
58	Various	121.6	122.2
59	4403	114.0	114.0
60	4405	103.4	103.8

GOTHIC Control Volume	Reactor Building Rooms/Areas Included	Maximum Vapor Temperature (F)-- CLTP	Maximum Vapor Temperature (F)-- CPPU
61	4406	121.8	122.4
62	4407	122.0	122.5
63	4412	103.5	103.7
64	4413	112.7	113.1
65	4502, 4620	120.0	122.6
66	4503, 4621	121.8	122.4
67	4506	110.4	110.8
68	4510	129.0	129.5
69	4511	126.8	127.4
70	4512	128.7	130.3
71	4606	116.2	116.6
72	4609	109.4	110.4
73	4613	130.4	130.6

13.2 In the Hope Creek PUSAR, Section 4.4, you addressed the Main Control Room (MCR) Atmosphere Control System. It was stated that "there are no changes to the MCR envelope and there are no significant changes to the temperatures in the adjacent walls and ceilings." Describe the areas surrounding the Control Room and what was considered in those areas to conclude that there are no significant changes to the temperatures in the adjacent walls and ceilings of the Control Room.

Response

As stated in NEDC-33076P, Section 4.4:

"Heat sources in the main control room (MCR) are due to equipment, ambient outside air temperature, and personnel and do not change with CPPU. There are no changes to the MCR envelope and there are no significant changes to the temperatures in the adjacent walls and ceilings."

Temperature changes in the Control Building are negligible. CPPU does not add any electrical or electronic equipment to the Control Room or Control Building. CPPU may add some amperage for control and indication signals but the resulting changes in temperatures are considered negligible. The rooms that are adjacent to the control room contain ventilation equipment or electrical and electronic equipment. These rooms are not impacted by CPPU and therefore there are no temperature changes in these areas that can impact main control room temperatures.

Reactor Building heat loads (adjacent to the Control Building) after EPU implementation are documented in PSEG calculation GR-0001, Revision 7. The revision to the calculation incorporates slight changes in heat loads due to maximum ECCS suction piping temperature increases due to EPU conditions. The slight increase is bounding for all conditions up to and including a rated thermal power of 3,952 MWt.

13.3 Discuss and confirm that the Filtration, Recirculation, and Ventilation System's (FRVS) ability on achieving a negative draw down pressure in the secondary containment is not impacted by the EPU. Also, identify the maximum FRVS inlet temperature under EPU operating conditions and its relationship to any design inlet temperature limitations.

Response
The FRVS capability to meet design requirements (including RG 1.183) after EPU implementation has been evaluated and documented by PSE&G calculation 11-0066, Rev. 7 "FRVS Drawdown and Long-Term Post-LOCA Reactor Building Temperatures", and calculation 11-0079, Rev. 2 "FRVS Drawdown & Long-Term Heat Loads".

Calculation 11-0079 utilizes the reactor building heat loads present in the short-term following a LOCA/LOP scenario. The results are used as input into the FRVS drawdown calculation 11-0066 to ensure the FRVS vent fan(s) can drawdown the reactor building pressure to –0.25" within the Technical Specification 4.6.5.1.c.1 required 375 seconds. The calculated drawdown time at CLTP was 221 seconds. The calculated drawdown time after EPU is 238 seconds. The increase in FRVS drawdown is attributed to increased reactor building temperatures that occur within the first 375 seconds following a DBA LOCA.

Calculation 11-0066 uses GOTHIC software to analyze the reactor building drawdown and to predict long-term post LOCA reactor building room temperatures for EPU conditions. As tabulated in the response to RAI 13.1, all area temperatures within the reactor building are below 131°F. The FRVS limiting component design temperature is 175°F. The maximum calculated area temperature of 131°F is below the 175°F FRVS temperature limit and well below the charcoal ignition temperature of 625°F.

13.4 In the Hope Creek EPU application request, Section 6.6 of Attachment 6, you addressed Diesel Generator Room (SDG) temperature. It was stated that the SDG remains below rated capacity. Confirm that the design basis heat loads are based on rated capacity (not actual loading) and assure that the ability of the safety-related SDG Room Recirculation System coolers to maintain the room within the required temperature is not impacted by the EPU.

Response
EDG Room design basis heat loads are based on full rated capacity of the EDG. Each EDG room contains two room coolers that are supplied with cooling from separate SACS loops. These coolers were designed to be 100% coolers, either of which can maintain room temperature within design limits at full rated capacity of the EDG.

EPU has no impact on EDG room cooling. There are no process temperature changes in the area and there are no changes in EDG room heat loads at EPU.

13.5 In the Hope Creek EPU application request Section 6.6 of Attachment 6 states that there is no increase in the design basis heat loads in the SFP area. Discuss and confirm that the effects on the SFP area Heating Ventilation and Air Conditioning (HVAC) system due to higher burnup fuel in the spent fuel pool are fully considered. Also, address whether there are any effects due to EPU on the ventilation system that could result from loss of SFP cooling.

Response

The Reactor Building Ventilation System (RBVS) has three 50% capacity supply and exhaust trains available for refueling operations. A refueling floor exhaust duct high radiation signal results in the auto shutdown of RBVS, isolating the flow path to the environment by closing RBVS dampers and auto starting the Filtration, Recirculation and Ventilation System (FRVS) system. The FRVS recirculation system is an engineered safeguards HVAC system that is connected in parallel with RBVS supply and exhaust ductwork. The FRVS recirculation system consists of six 25%-capacity air handling units rated at 30,000 cfm each. This system circulates reactor building air through filters and cooling coils during a LOCA and uncontrolled releases to the environment. A small volume of air is diverted from the FRVS recirculation exhaust duct by the FRVS vent system and released to the environment via the plant vent after passing through HEPA and charcoal filters located in the FRVS system.

Hope Creek (HCGS) HVAC systems were evaluated at a CPPU power level of 3952 MWt which bounds the proposed CPPU of 3840 MWt. The results of the analyses indicate the HVAC systems have sufficient margin and no changes are required. No changes in refueling floor HVAC loading result from CPPU because the existing licensing basis SFP temperatures are maintained by the Fuel Pool Cooling and Cleanup System. HCGS can maintain SFP pool temperature limits for offloads below the current licensing limits of 135°F (batch) and 150°F (full core) with increased CPPU decay heat loads.

The refueling outage planning process requires that decay heat loads to be transferred to the SFP combined with the existing load in the SFP are evaluated with respect to actual plant conditions (i.e. SFP starting temperature, SACS cooling water temperature, available flow rates, etc) to assure SFP temperature limits are not exceed after core offload (PSE&G calculation EC-0074). The use of actual plant conditions in performing EC-0074 provides conservative time limits for contingency actions should a loss of a SFP cooling component occur.

The available options to restore and maintain SFP cooling include placing in service safety-related FPCC system pumps (two available), FPCC heat exchangers (two available), the RHR assist mode or Alternate RHR assist mode that allows use of an RHR pump and RHR heat exchanger. These options do not change as a result of CPPU. This level of safety-related redundancy provides multiple options to maintain SFP temperatures within design limits should loss of a SFP cooling component occur. A complete loss of SFP cooling would require multiple simultaneous failures of safety-related pumps and safety-related heat exchangers. Therefore, complete loss of SFP cooling is highly unlikely.

Contingency planning for a loss of a SFP cooling component includes manually placing the FRVS recirculation system in-service on a loss of SFP cooling. This is a pre-cautionary step to prevent a release should airborne particulate levels increase before SFP cooling is restored. This requirement is not changed by CPPU. A failure to implement contingency plans within the calculated time frames established by EC-0074 could result in SFP boiling. In the unlikely event that pool boiling would occur, the consequences of a boiling SFP after implementation of CPPU are not significantly different than with a boiling pool at CLTP.

13.6 Are there any modifications planned to the HVAC systems (including atmospheric cleanup systems) as a result of the EPU? Clearly define the areas that will see higher heat loads due to EPU, magnitude of the increase, and the basis for determining that the existing systems are adequate under post EPU conditions (with or without modifications).

Response
No modifications are planned to any HVAC or atmospheric cleanup systems. The EPU impact on HVAC systems are minimal as further described below:

System	EPU Impact
Control Building Ventilation	Negligible impact due to negligible increase in control and indication signals. No heat-generating equipment added to Control Building.
EDG Ventilation	No impact; HVAC sized for EDG full rated load.
Radwaste Building Ventilation	Negligible impact. Small increase in radwaste volume but no significant temperature changes.
Drywell Cooling	Feed water temperature leaving FWH #6 increases by 15.2°F. This results in a negligible increase of approximate 0.2°F in Drywell temperature.
Reactor Building HVAC/Steam Tunnel	Feed water temperature leaving FWH #6 increases by 15.2°F. This results in a negligible increase of approximate 0.5°F in the steam tunnel area.
Reactor Building HVAC/ECCS Rooms	RHR and CS pump rooms and RHR HX rooms increase by approximately 0.2°F due to suppression pool temperature increase. No impact on HPCI, RCIC, or SACS pump rooms.
Reactor Building/SFP Area	No impact
Reactor Building/TSC Area	No impact

System	EPU Impact
Turbine Building	General TB areas were designed with an approximate 15% heat load margin. The largest EPU heat load increase in TB general areas is estimated at about 11%. Hence no significant temperature increases are anticipated. Other specific TB temperature changes include: • Moisture separator rooms = 1.1°F increase • Feed water pump area = 2.0°F increase • Feed water heater #6 area = 2.0°F increase • Lower FWH areas (#3, 4, 5) = 3.5°F increase • Condensate pump rooms = negligible increases • Steam tunnel area = 0.5°F increase
Turbine Building Recirculation MG Set Area	No impact

13.7 In the Hope Creek PUSAR, Section 4.1, explain why the choice of the Residual Heat Removal (RHR) heat exchanger "K" value is conservative. Describe the program to ensure that the actual value is not less than this value.

Response

PUSAR paragraph 4.1.1.1 shows that the assumed RHR heat exchanger performance is based on heat exchanger efficiency (K-value) of 307 Btu/sec-°F. This is the same K-value that was derived for the RHR heat exchanger in the late 1990s when maximum SACS temperature was increased to 100°F and is used in the current HCGS containment response analysis of GE-NE-T2300759-00-02. The K-value is not changed for CPPU. The K-value of 307 Btu/sec °F was accepted and discussed by the NRC in the SER to Amendment 120 dated 4/19/1999. In this SER, the staff found that use of the GE SHEX code to establish the heat exchanger K-value to be acceptable. In the SER for Amendment 120, the Staff also found that the K-value had been determined in a maximum fouled condition.

RHR heat exchanger loads change due to EPU during design basis accident conditions (LOCA and LOP) due to increased predicted suppression pool temperatures. The post-LOCA suppression pool temperature increases to 212.3°F and post-LOP to 213.6°F assuming operation at a RTP of 3840 MWt. The RHR heat load during a LOCA increases from 121.7 (CLTP) to 127.1 MBtu/hr (CPPU). The 127.1 MBtu/hr value was derived from a maximum suppression pool temperature of 215.6°F, which was based on an assumed RTP of 102% of 3952 MWt. Therefore, the actual RHR heat exchanger heat load would be less at 3840 MWt. Consequently, heat exchanger margins exist in both the assumed heat exchanger fouling during the derivation of the K-value and also in the LOCA analysis that was performed at an RTP of 102% of 3952 MWt (4031 MWt) rather than 3840 MWt (a 5% difference).

The RHR heat exchangers at Hope Creek contain demineralized water in both the shell (RHR) and tubes (SACS). As such, these heat exchangers are not susceptible to fouling, silting, grassing, or related degradation mechanisms as would be the case with raw water systems. Since these heat exchangers contain demineralized water, the K-value is assured if the required flow rates (RHR and SACS) are periodically confirmed. These flow rates are confirmed every 18-months by plant surveillance procedures.

13.8 In the Hope Creek PUSAR, Section 4.1, verify that all input parameters to the containment peak pressure and temperature, environmental qualification and subcompartment analyses remain the same as those in the updated final safety analysis report except for those affected by the power uprate. For example, containment volume, heat sink descriptions, heat exchanger performance, equipment flow rates and flow temperatures, initial relative humidity, ultimate heat sink temperature, etc. justify any changes made for the power uprate analyses.

Response

Comparisons of various key input parameters in the HCGS UFSAR and in EPU analyses are shown below. The containment pressure and temperature analyses bound the equipment qualification pressure and temperature profile, therefore they are both shown in the following table.

Containment Response/EQ:

Input Parameters	**HCGS UFSAR**	**CPPU**
Reactor Thermal Power	3339 MWt	3840 MWt
Average Coolant Pressure	1020 psia	1020 psia
Volume of Liquid in Vessel	11,885 ft^3	11,716 ft^3
Containment Design Pressure	62 psig	62 psig
Drywell Design Temperature	340°F	340°F
Wet Well Design Temperature	310°F	310°F
Drywell Net Free Volume	169,000 ft^3	169,000 ft^3
Suppression Pool Low Water Volume	118,000 ft^3	118,000 ft^3
Initial Suppression Pool Temperature	95°F	95°F
RHR Heat Exchanger Performance	307 Btu-Sec °F	307 Btu-Sec °F
RHR Pool Cooling Flow Rate	10,000 gpm	10,000 gpm
Drywell Spray Flow Rate	9500 gpm	9500 gpm
Initiation of RHR	600 seconds	600 seconds
Initial Containment Pressure	1.5 psig	0 to 1.5 psig
Initial Containment Temperature	135°F	135°F
Initial Relative Humidity	20%	20%
Initial Wet Well Relative Humidity	100%	100%
UHS Temperature	89°F	89°F
SACS Temperature	100°F	100°F
MSIV Closure Time (RSLB)	3.5 sec	3.5 sec
Drywell Passive Heat Sinks		
Drywell Shell	0 ft^2	17,850 ft^2
LOCA Vent System	0 ft^2	15,256 ft^2
Wet well Passive Heat Sinks		

Upper Torus Shell	0 ft^2	15,408 ft^2
Lower Torus Shell	0 ft^2	15,408 ft^2

As can be seen above, the input parameters for containment peak temperatures and pressures are essentially unchanged for CPPU. Passive heat sinks, both in the drywell and wetwell are input to the analysis as described below. Passive heat sinks were not modeled for previous containment response analyses.

The passive heat sinks are credited in the Hope Creek EPU Containment Analyses for certain events, in order to show that adequate margin to the acceptance limits are available at EPU conditions. The heat sinks modeled include the drywell metal shell, the vent system metal, and the torus metal shell,. Both the submerged portion of the torus shell in contact with the suppression pool water and the torus shell portion in contact with the torus airspace are modeled.

The events analyzed are:

- Long-term DBA-LOCA with the RHR system in suppression pool cooling mode (UFSAR Case C), and
- Loss of Offsite Power (LOP).

In all of the above cases, the heat slab modeling capability of the SHEX code is used, with the applicable physical and thermal hydraulic parameters represented accordingly for the passive heat sinks in the drywell, the wetwell gas space and the suppression pool.

The SHEX analyses for all cases shows that a detailed, mechanistic heat slab model is used, which takes into account the material properties, geometry, and interfacial heat transfer coefficients between the gas spaces and the heat slabs and between the suppression pool and the heat slabs. Heat transfer from the heat sinks to the reactor building is conservatively neglected.

The results show that crediting heat sinks results in a 2°F decrease in the peak suppression pool temperature, which is judged to be reasonable. Crediting heat sinks using the methodology described is therefore reasonable and justified.

Sub-Compartment Analysis (RSLB)

Input Parameter	**HCGS UFSAR**	**CPPU**
Limiting Annulus Line Break	Recirculation Suction Line Break	Recirculation Suction Line Break
Flow Split (Containment/Annulus)	50%/50%	75%/25%
RPV Dome Pressure	1020 psia	1020 psia
Reactor Power (CLTP)/Pump Speed	100%/100%	66.2%/39.2%
Enthalpy	544.4 Btu/lbm	498.8 Btu/lbm

In the case of the sub-compartment analysis, the change from 100% power/100% flow to 66% power/40% flow was made for both the MELLLA and CPPU conditions because it causes greater (and therefore more conservative) mass-and-energy release due to the lower enthalpy. Hence this is an approved methodology change implemented to provide the bounding analysis and is more conservative than the previous analysis at CLTP.

The CPPU sub-compartment analysis flow-split changes from 50%/50% to 75%/25% because a flow restrictor was installed in the shield wall subsequent to the initial sub-compartment analysis. The resultant forces calculated in the CPPU analysis are bounded by the forces calculated in the original analysis. The 75%/25% split is part of the Hope Creek current licensing basis for CLTP, but the UFSAR continues to reference the analysis based on a 50%/50% split because it provided loads for design of the shield wall and reactor vessel. The bounding analysis (using the 50%/50% flow split) for the resulting forces will continue to be documented in the UFSAR after CPPU. These forces will remain in the UFSAR because they are the design forces for the structures and because they continue to bound current conditions. Additional discussion on annulus pressurization loads is provided in the response to RAI question 13.11.

13.9 In the Hope Creek PUSAR, Section 4.1, what is the temperature limit for piping attached to the torus? What is the calculated peak temperature of this piping?

Response

Torus attached piping (TAP) has been evaluated for OLTP conditions and is documented in volume 6 of the NUTECH Plant Unique Analysis Report BPC-0-300-1, Revision 0, dated January 1984. Although a specific temperature is not listed for each TAP system for both internal and external torus piping in the report. The TAP piping have different temperatures depending on the system. The maximum temperatures for various systems attached to the torus are given below from Specification P-501.

- RHR 188 to 470 deg.
- RCIC 140 to 170 deg.
- Core Spray 212 deg.
- HPCI 140 to 212 deg.
- Torus Water clean up 150 deg.
- RCIC Turbine Steam 190 to 267 deg.
- HPCI Turbine Steam 366 deg.
- Containment Atmosphere control 340 deg.

Changes in peak TAP temperature due to CPPU are attributed to increased suppression pool temperature. Peak suppression pool temperature can be used to infer the peak temperature of the TAP. The calculated CPPU peak suppression pool temperature for containment response is 213.6°F at 102% of 3840 MWt. It is noted that this temperature is derived by applying an SRV blowdown strategy to achieve a maximum suppression pool temperature. The suppression pool temperature following the DBA LOCA is 212.3°F at 102% of 3840 MWt. For the systems which are analyzed at 212°F, the small temperature increase will have an insignificant effect on piping and support.

13.10 In the Hope Creek PUSAR, Section 4.1.1.2, containment structural design basis temperature is stated to be 340 °F. This is higher than that of some other BWRs and is usually the temperature limit for Environmental Qualification (EQ). Verify that 340 °F is the correct value.

Response
The Hope Creek containment design basis temperature is 340°F. The drywell maximum design temperature of 340°F is tabulated in Table 1-2.2-1 on page 1-2.19 of the Plant Unique Analysis Report BPC-0-300-1, Revision 0, dated January 1984. The 340°F design value is also tabulated in Table 3 of the Hope Creek Generating Station Environmental Design Criteria, Calculation No. D7.5 Revision 19, dated January 15, 2003, and in UFSAR Section 6.2.1.1.3 (Containment Functional Design). CPPU does not change this temperature limit.

13.11 In the Hope Creek PUSAR, Section 4.1.2.3, provide the value of pressure differential calculated for the EPU and the Hope Creek pressure difference limit.

Response
Recirculation Suction Line Break (RSLB) and Feed Water Line Break (FWLB) cases evaluated for ARTS/MELLLA bound the CPPU mass and energy releases for annulus pressurization (AP) and reactor pressure vessel (RPV) loads. The ARTS/MELLLA analyses have been reported to the NRC in LCR H04-01 (ARTS/MELLLA Implementation), with specific details on AP loads included in response to NRC requests-for-information (RAI) by HCGS letter LR-N05-0213 (Reference 13.11-1). Since the ARTS/MELLLA cases bound CPPU, the descriptions and information in LR-N05-0213 remain applicable to CPPU.

The information of LR-N05-0213 contains several plots of pressures and forces from the various analyses along with detailed descriptions on the flow restrictor that was installed during plant construction to reduce mass and energy release into the annulus area. For convenience, a brief summary of HCGS annulus pressurization loads is provided below.

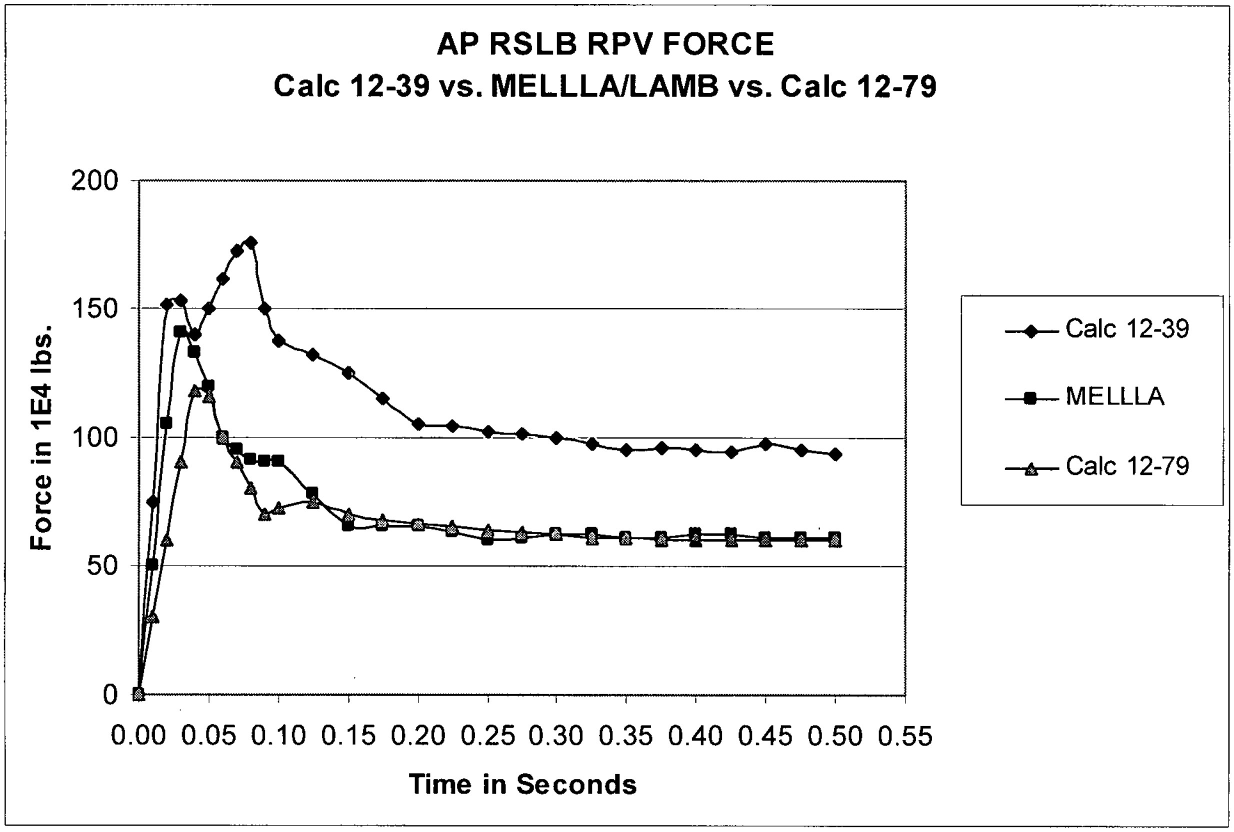
Annulus pressurization loads, both on the reactor pressure vessel (RPV) and the biological shield wall (BSW) are derived by converting time-dependent mass-and-energy release data into loads or forces (in thousands of pounds) on the various components. Original design basis calculations of these loads were performed with the conservative assumption that 50% of the mass and energy release would discharge directly to the containment, with the other 50% going to the annulus area. These calculations provided the loads on which the designs of both the RPV and shield wall were based.

These original design loads were based on 100% power/100% flow conditions in the RCS. Since original design, it has been determined that the minimum recirculation pump speed point on the MELLLA load line is more limiting (than 100%/100%) for recirculation suction line breaks (RSLB) and feed water line breaks (FWLB). This methodology change (which changes the enthalpy of the water/steam released through the break) has a much greater impact on both CLTP loads (with the MELLLA load line) and CPPU loads than the impact of the increased power level. The results of the methodology change were submitted to the NRC with the MELLLA LCR.

The methodology change notwithstanding, the original design calculations continue to bound both CLTP (with minimum pump speed) and CPPU (with minimum pump speed) conditions because subsequent to the original design, a flow restrictor was installed in annulus penetration that diverts 75% of the mass-and-energy directly to the containment in the event of a line break, with only 25% to the annulus. The 75%/25% flow split is sufficient to compensate for the additional energy provided by the enthalpy change and, the case of CPPU, by the enthalpy change and the higher power level.

The following figure compares RPV loads (in KIPS) from the original design basis calculation with current analyses. The specific plots are described below:

- Calculation 12-39 is the original design basis calculation that assumes the 50%/50% flow split. This calculation remains the plant design basis and the results of this calculation are currently shown in UFSAR Figure 6B-8.
- Calculation 12-79 is the CLTP calculation with 75%/25% flow split and 100%power/100% flow.
- MELLLA is CLTP analysis at with the 75%/25% flow split and the 100% power/minimum pump speed. This analysis bounds CPPU loads



As stated above, additional plots and information on annulus pressurization loads are contained in HCGS letter LR-N05-0213, which were submitted to the NRC with LCR H04-01 (MELLLA Implementation).

<u>Reference</u>

13.11-1 PSEG letter LR-N05-0213, Response to Request for Additional Information: Request for Change to Technical Specifications: ARTS/MELLLA Implementation, May 20, 2005

13.12 In the Hope Creek PUSAR, Table 1-1 shows that both the "STEMP" and the "SHEX" codes are used for the Anticipated Transient Without Scram (ATWS) event. Describe the function of each code in this calculation.

<u>Response</u>

The STEMP code is used to determine the maximum suppression pool temperature and the maximum containment pressure during ATWS through the time of Reactor Hot Shutdown without vessel depressurization. The SRV steam flow is calculated by the ODYN code and input to STEMP.

The SHEX code is used to calculate the suppression pool temperature response from the start of the event through the time vessel depressurization is completed.

In the SHEX calculation, the SRV steam discharge to the suppression pool calculated by ODYN is used to the time of hot shutdown. The SRV steam discharge after hot shutdown, during vessel depressurization, is calculated internally by the SHEX code.

13.13 In reference to the Hope Creek PUSAR, Section 4.1, is the metal-water reaction increased by the EPU? What is the effect on containment response?

Response
The metal-water reaction energy vs. time relationship is calculated using the method described in USNRC Regulatory Guide 1.7 as a normalized value (fraction of reactor thermal power). All of the energy from the metal-water reaction is assumed to be transferred to the reactor coolant in the first 120 seconds into the LOCA. Thus, it is scaled up proportionately for EPU. The metal-water reaction energy represents a very small fraction of the total shutdown energy transferred to the coolant. Therefore, it is concluded that the effect of the increase in metal-water reaction energy on containment response as a result of the EPU is negligible.

13.14 In reference to the Hope Creek PUSAR, Section 4.1.1.1(a), please provide the peak suppression pool temperatures resulting from the postulated ATWS, Station Blackout and 10 CFR Part 50, Appendix R Fire events.

Response
The following table lists peak suppression pool temperatures under various events or conditions as evaluated for EPU power levels.

Event	Peak Suppression Pool Temperature	Evaluation Power Level
Limiting ATWS (Pressure Regulator Failed Open)	199.0°F	3952 MWt
Station Blackout	198.0°F	3952 MWt
Appendix R Fire	205.9°F	3840 MWt

13.15 In reference to the Hope Creek PUSAR, Section 4.7, what, if any, changes are necessary to Containment Atmosphere Deluge System, CADS operation or nitrogen storage due to the power uprate?

Response
Hope Creek maintains an inerted containment atmosphere in accordance with the requirements of 10 CFR 50.44(b)(2). Containment and reactor building temperatures during normal operation are not significantly affected by CPPU. Consequently, the vacuum relief and nitrogen inerting functions of the Containment Atmosphere Control System (CACS) are not impacted by CPPU.

Accordingly, no changes to CACS or nitrogen storage are required for CPPU. Hope Creek does not rely upon a CAD System to maintain post- DBA LOCA combustible gas concentrations within the primary containment at or below the flammability limits.

Revised Questions from (SBWB):

3.2 The NRC staff review of previous EPU applications included evaluation of the dynamic effects and missiles that might result from plant equipment failures at EPU operating conditions, as well as the effects of a loss-of-coolant accident (LOCA). In the Hope Creek EPU application request, Section 2.8.6.2 of Attachment 11, does not address this specific concern, please justify why similar criteria does not apply to the Hope Creek spent fuel storage.

Response

The Spent Fuel Pool system is located in the reinforced concrete reactor building. Dynamic effects and missiles that might result from plant equipment failures have not changed with respect the plant's current design basis as discussed in the UFSAR section 3.5. HELB and MELB have been evaluated in PUSAR section 10.1, and 10.2. The resulting pressures and temperatures were found to be within current licensing values.

The frequency of catastrophic failure of rotating equipment having synchronous motors as discussed in the UFSAR is not probable. The conclusions in the current UFSAR section 3.5.1.1.1 have not changed.

Equipment important to safety will continue to be protected from the effects of turbine missiles as discussed in the notes to section 2.5.1.2.2 in Attachment 10 to Reference 3.2-1. HCGS HP and LP turbine rotors have been converted to the monoblock design and both the "normal overspeed" and "emergency overspeed" trip values were confirmed. The Low Pressure (LP) Turbine rotors at Hope Creek have now been converted to the monoblock design from the original built-up design that was installed with the original construction of the plant. This LP rotor conversion from built-up to the monoblock design effectively increased total rotor inertia values by more than 21% over the original rotors. This large increase in inertia slows the accelerations rate of the machine should a load rejection event occur. For the same conversion, the flow increases enabling the power output of the machine to increase almost 20% over the original power levels. From an overspeed standpoint, these two changes effectively cancel each other out, such that the estimated peak speed following a full load rejection remains virtually unchanged from its original estimated value of 109.26% of rated speed. The latest estimate following the LP conversion is, 109.20% of rated speed. GE refers to this estimated peak speed as "normal overspeed" or, NOS. For NOS, it is assumed that all protective steam valves and control systems have responded as intended to minimize the resulting peak speed.

Consequently, there is no need to adjust the design setting of the mechanical trip, which remains at 109.9 - 110.4% of rated speed, as there is still sufficient margin between the NOS value and the minimum mechanical trip setting. This margin should normally be at least 0.5%, and presently, it is 0.7%.

Reference

3.2-1 PSEG letter LR-N05-0258, Request for License Amendment: Extended Power Uprate, November 7, 2005

3.5 In NEDC-33172P, "SAFER/GESTR-LOCA for HCGS at Power Uprate," it was reported that the Licensing Basis PCTs are 1380 F for GE14 and 1540 F for SVEA-96+. Please provide the following additional information:

a) What is the corresponding break size for above Licensing Basis PCTs, and is it classified as small-break or large-break LOCA? Is the current Licensing Basis PCT is based on small-break or large-break LOCA? If they are different, explain why.

Response

The EPU Licensing Basis PCT for both the GE14 and SVEA fuel is based on the DBA large break LOCA. This is unchanged from the current CLTP Licensing Basis PCT. The corresponding break size for the DBA large break is 4.085 ft^2.

b) Was top-peaked and mid-peaked axial power shape included in establishing the MAPLHGR and determining the limiting PCT?

Response

A mid-peaked power shape was assumed for both large break and small break LOCA events. A discussion of the potential impact of axial power shape on LOCA PCT is provided in the response to question 3.47.

c) In previous EPU LOCA analyses, the NRC staff has noted that the fuel types did not significantly impact the value of PCT, provided that the limiting LOCA event was a small break. The explanation for this was based on the fact that the affect of fuel stored energy was insignificant for small-break LOCA. Explain why a relatively large difference in PCT values (160 °F) between GE14 and SVEA-96+ exists.

Response

The Licensing Basis PCT for both the GE14 and SVEA fuel is based on the DBA large break LOCA. Therefore, the explanation that the limiting PCT is not significantly impacted by the fuel type, based on the small break event being limiting, does not apply. A discussion of the difference between the GE14 and SVEA large break PCT results is provided in the response to question 3.47.d below.

d) The PUSAR indicates that the limiting PCT for GE14 increases 10 °F from 1370 °F to 1380 °F before and after EPU. Please provide the limiting PCT for SVEA before and after EPU. The PCT changes due to EPU were typically within 20 °F. Please confirm if it is also true for SVEA fuel. If not, then please explain why.

Response
The Licensing Basis PCT for the SVEA fuel is 1540°F at both CLTP and EPU power, and is based on the same power / flow conditions, i.e., MELLLA flow at CLTP Appendix K power (3430 MWt at 76.6% of rated core flow). [[

]]

New Question from (SBWB):

3.57 Section 1.2.2 Computer Codes, Table 1-2 of the PUSAR lists all the nuclear steam system codes used for the EPU request. This section indicates that the HCGS application of these codes complies with the limitations, restrictions, and conditions specified in the applicable NRC safety evaluation report that approved each code, with exceptions as noted in Table 1-2. The NRC staff has noted that in Section 2.0 of Attachment 15 to the submittal, a limited number of those codes (TGBLA, PANACEA, ISCOR, ODYN, TASC, SAFER and GESTR), and their methods and range of applications were discussed. However, the report did not include all the codes listed in Table 1-2 of the PUSAR.

Please review the fuel vendor's analytical methods and code systems (neutronic, LOCA, transient, and accidents, etc.) used to perform the safety analyses supporting the HCGS EPU application and provide the following information:

a) Confirm that the steady state and transient neutronic and thermal-hydraulic analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges.

b) Confirm that for the EPU conditions, the calculational and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions.

c) Confirm that the assessment database and the assessed uncertainty of models used in all licensing codes that interface with or are used to simulate the response of HCGS during steady state, transient or accident conditions remain valid and applicable for the EPU conditions.

<u>Response</u>
Attachment 15 of the PSEG EPU submittal is a plant-specific supplement to the GE submitted Licensing Topical Report (LTR) NEDC-33713P, "Applicability of GE Methods to Expanded Operating Domains". The PSEG EPU submittal for HCGS has referenced NEDC-33713P as the basis for the applicability of the GE methods to EPU. The Attachment 15 supplement provides additional plant-specific information based on a preliminary EPU core design for Cycle 15.

Table 3.57-1 below identifies which codes used in the HCGS EPU analysis are addressed by NEDC-33173P and provides a basis for the acceptability of the remaining codes.

Table 3.57-1		
Task	**Computer Code**	**Impact of EPU**
Nominal Reactor Heat Balance	ISCOR	This simplistic application is not impacted by EPU.
Reactor Core and Fuel Performance	TGBLA PANACEA GESAM	Covered by NEDC-33173P. Covered by NEDC-33173P. Covered by NEDC-33173P.
Reactor Power/Flow Map	None	See Note 1.
Thermal Hydraulic Stability	ODYSY PANACEA ISCOR TRACG	Covered by NEDC-33713P. Covered by NEDC-33713P. See Note 8. Covered by NEDC-33713P. See Note 8. TRACG addressed in NEDO-32465-A. See Note 2.
Reactor Vessel Fluence	DORTG01 TGBLA	See Note 3. See Note 3.
Reactor Internal Pressure Differences	ISCOR LAMB TRACG	Covered by NEDC-33173P. See Note 4. Application is different than described in NEDC-33173P. See Note 4.
Containment System Response	SHEX M3CPT LAMB	See Note 4. See Note 4. See Note 4.

Table 3.57-1		
Task	Computer Code	Impact of EPU
Transient Analysis	PANACEA ISCOR ODYN SAFER TASC	Covered by NEDC-33173P. Covered by NEDC-33173P. Covered by NEDC-33173P. Covered by NEDC-33173P. Covered by NEDC-33173P.
Anticipated Transient Without Scram	ODYN STEMP PANACEA ISCOR TASC SHEX	Covered by NEDC-33173P. See Note 4. Covered by NEDC-33173P. Covered by NEDC-33173P. See Note 5. See Note 4.
Station Blackout	SHEX	See Note 4.
Appendix R Fire Protection	GESTR SAFER SHEX	Covered by NEDC-33173P. Covered by NEDC-33173P. See Note 4.
Reactor Recirculation System	BILBO	See Note 6
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	See Note 4. Covered by NEDC-33173P. Covered by NEDC-33173P. Covered by NEDC-33173P. See Note 4.
Fission Product Inventory	ORIGEN2	See Note 7
High Energy Line Break	COMPARE-MOD 1	See Note 9

Note 1. Table 1-2 of the PUSAR lists BILBO as the computer code used to generate the Reactor Power/Flow Map. BILBO was not used to generate the reactor power/flow map and the reference to that code in Table 1-2 was an inadvertent oversight.

Note 2. For EPU application, the limiting power/flow point for the DIVOM application is the same as in the pre-EPU MELLLA application. GE had provided the 50.59 assessment report (GE-NE-0000-0052-5690-R0, TRACG04 DIVOM 10 CFR 50.59 Evaluation Basis, April 2006) to HCGS earlier.

Note 3. Vessel fluence calculation using DORTG01 is based on the NRC approved GE fluence methodology documented in the fluence LTR NEDC-32983P-A. In essence, inputs to the DORTG01 calculations include the neutron cross sections, atom densities, neutron source distribution, geometry model, and other inputs related to the discrete ordinates model. These inputs are prepared by the users and are not built in the code. The range of application for DORTG01 is not affected by the EPU. TGBLA is used only to prepare fuel-related inputs for fluence calculation (atom densities, neutron yields per fission, etc.).

Note 4. These models are applied to post-LOCA or SRV discharge related thermal hydraulics conditions. The thermal hydraulic conditions existing during the simulation have not extended beyond the qualification basis of these models when applied to EPU.

Note 5. This model is applied for post dryout PCT calculation. The thermal hydraulic conditions existing during the simulation have not extended beyond the qualification basis of these models when applied to EPU.

Note 6. BILBO is not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GE for "Level 2" application and is part of GE standard design process. The application of this code has been used in previous power uprate submittals. The BILBO code recirculation hydraulic parameters are based on plant specific hydraulic parameters not power or flow specific. Thus, different power levels and flow levels will not affect the analysis. This allows the code to be used at various power and flow conditions and is applicable to EPU conditions.

Note 7. ORIGEN2, developed by the Oak Ridge National Laboratory, is an industry standard code for fission product inventory calculation. Fuel exposure for EPU is within the exposure limit for the code application.

Note 8. Both PANACEA and ISCOR codes are needed to support the ODYSY and TRACG evaluations.

Note 9 Reactor Building (RWCU) sub-compartment analyses were performed using the COMPARE Mod 1 computer code. The COMPARE results, which were conducted at 120% OLTP (3952 MWt), demonstrated insignificant changes in resulting pressures and temperatures in the analyzed sub-compartments. In addition, as discussed in the response to RAI 8.1 (Reference 3.57-1), the changes in RWCU system operating conditions are very small.

<u>Reference</u>

3.57-1 PSEG letter LR-N07-0056, Response to Request for Additional Information, March 22, 2007

References

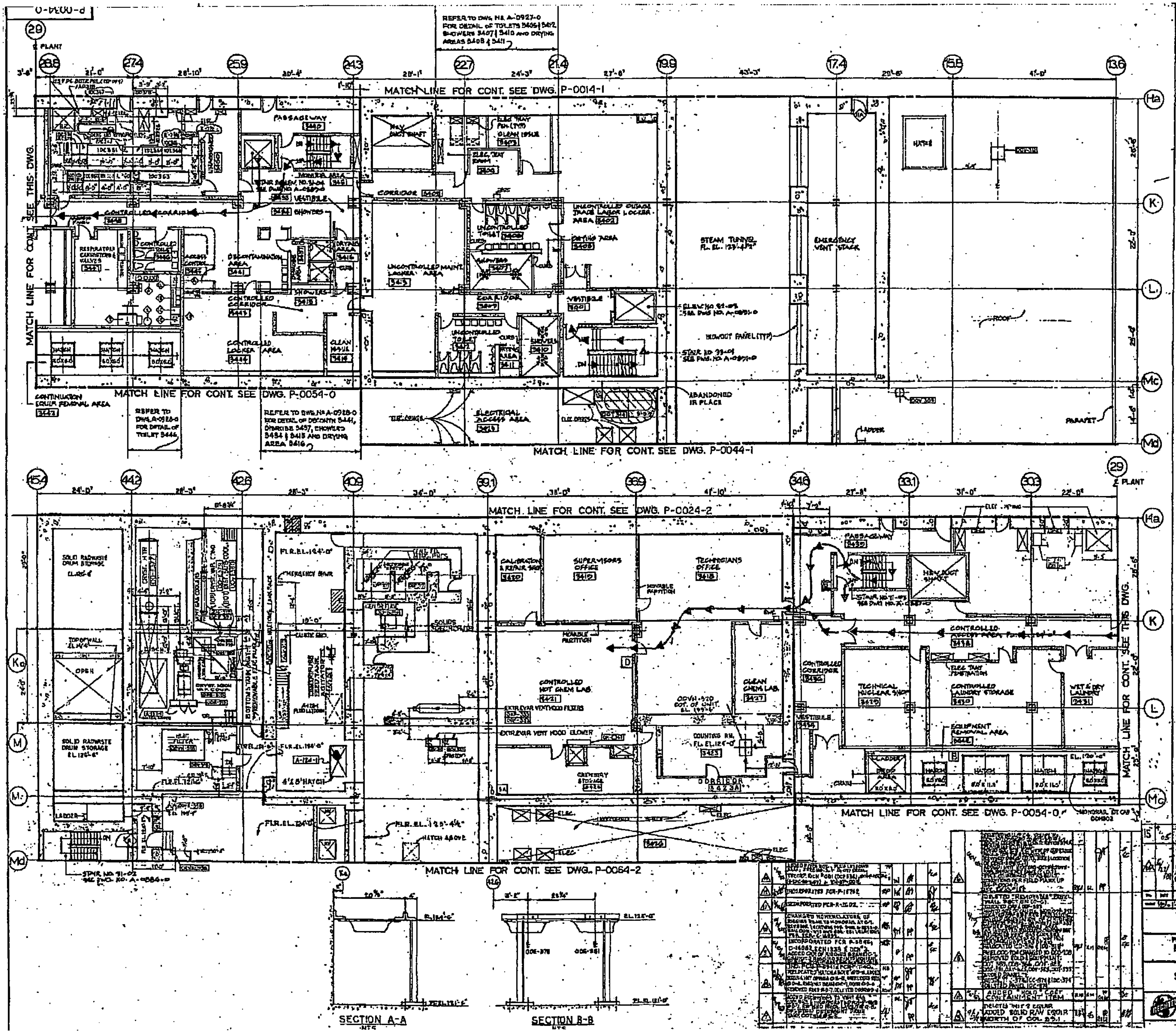
1. PSEG letter LR-N06-0286, Request for License Amendment: Extended Power Uprate, September 18, 2006

2. NRC letter, Hope Creek Generating Station - Request for Additional Information Regarding Request for Extended Power Uprate (TAC NO. MD3002), March 13, 2007

Plant Layout Drawings

RAI 11.7

P-0034-0
P-0035-0
P-0036-0
P-0053-0
P-0054-0
P-0055-0



KEY PLAN

EQUIPMENT SCHEDULE

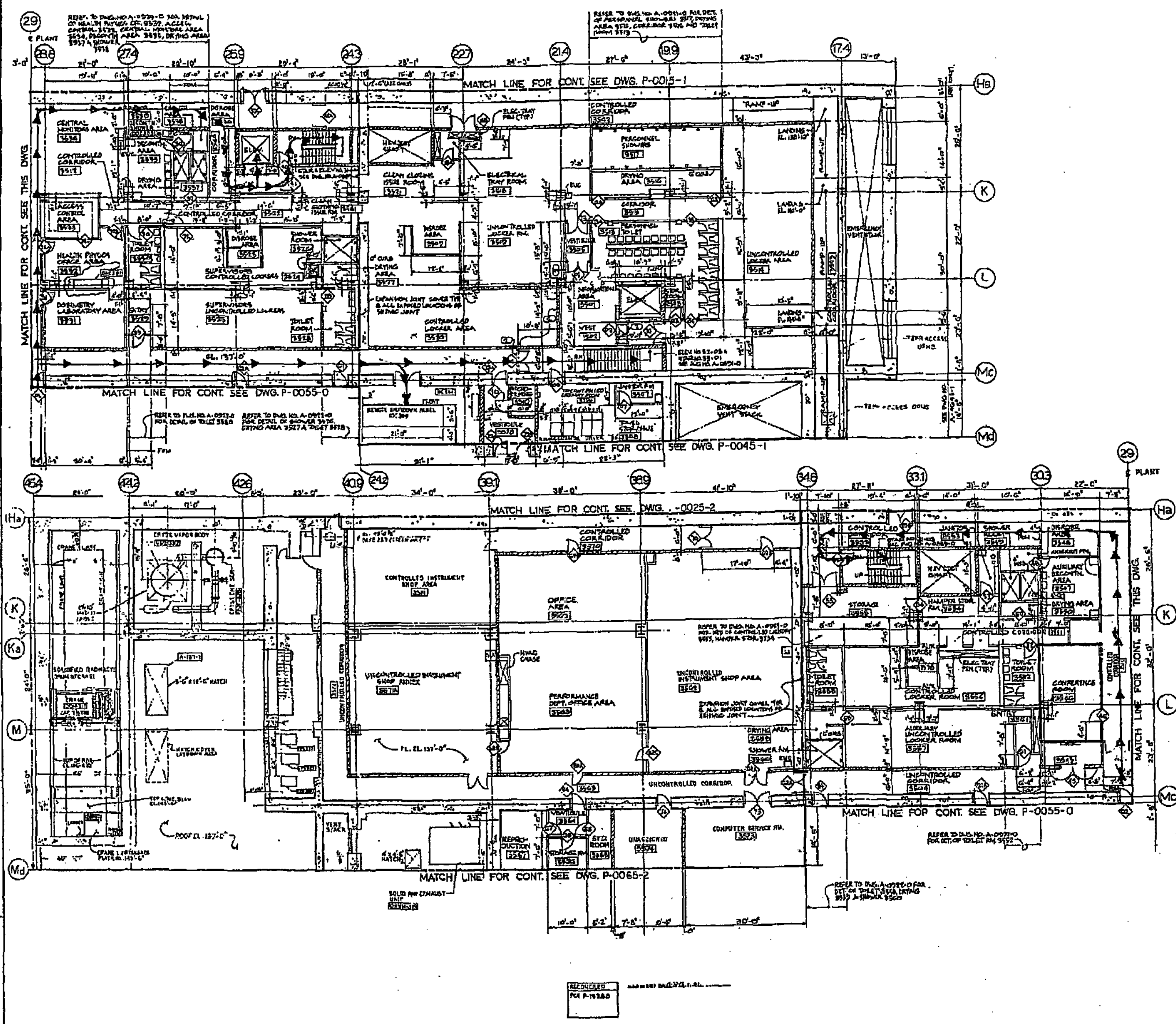
- (A) RECEIPT SURVEY TABLE
- (B) CAPTURE VELOCITY LOADING HO
- (C) WASHER
- (D) UNLOADING TABLE
- (E) VACUUM
- (2) ELECTRIC HOT WATER HEATER
- (3) WATER PROCESSING SYSTEM
- (4) HYDRO-PNEUMATIC EXPANSION TA
- (5) RESPIRATOR DRYER

BECHTEL
SAN FRANCISCO

PUBLIC SERVICE ELECTRIC AND GAS COMPA
HOPE CREEK GENERATING STATIO

EQUIPMENT LOCATION
SERVICE & RADWASTE ARE
PLAN EL. 124'-0"

P-0034-0



NOTES:
1. FOR ARCHITECTURAL WALL SYMBOLS SEE DWG A-0011-0

ATTENTION: ANY REVISION TO THIS DRAWING SHALL BE MADE ONLY BY CADD

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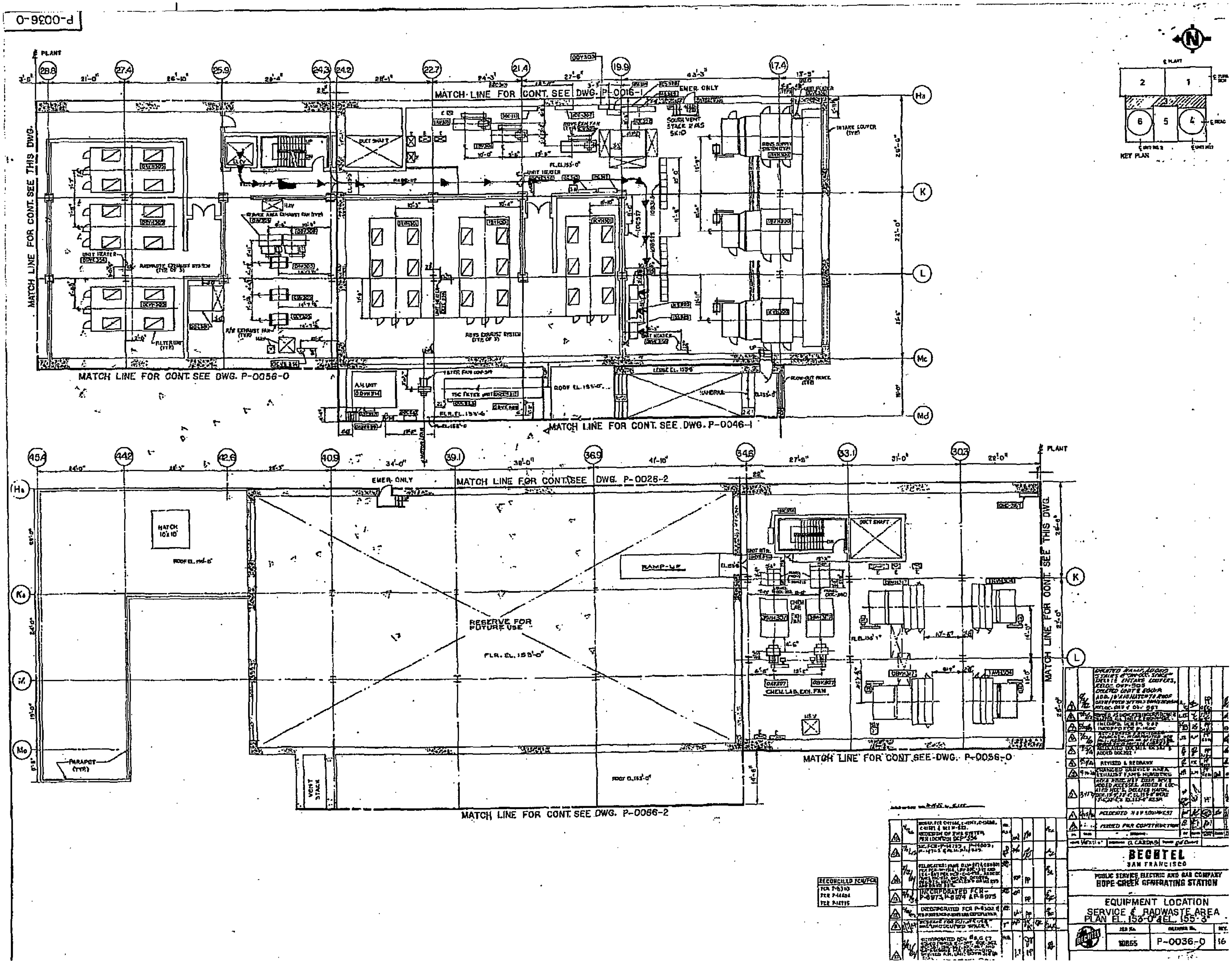
11	SEE NOTE "A" (B-2)	DR				

REVISION

HOPE CREEK GENERATING STATION
EQUIPMENT LOCATION
SERVICE & RADWASTE AREA
PLAN EL. 137'-0"

PSEG NUCLEAR, LLC

P-0035-0 -11



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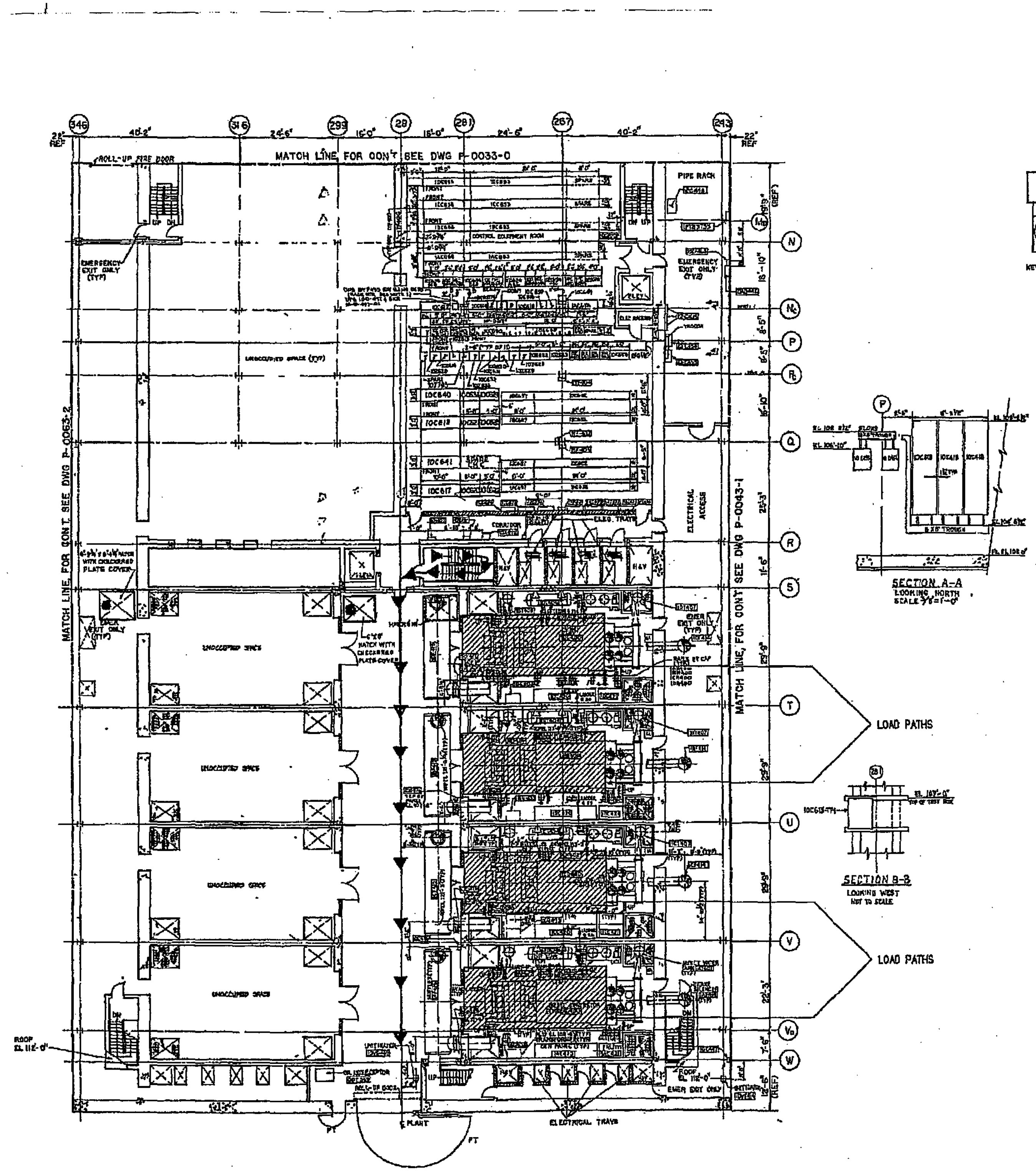
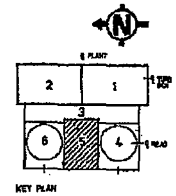
MATCH LINE FOR CONT. SEE DWG. P-0026-2
MATCH LINE FOR CONT. SEE DWG. P-0066-2
MATCH LINE FOR CONT. SEE THIS DWG.
MATCH LINE FOR CONT. SEE DWG. P-0056-0
EMER. ONLY
DUCT SHAFT
RESERVE FOR FUTURE USE
FLR. EL. 153'-0"
RAMP-UP
CHEM. LAB. EXH. FAN
VENT STACK
PARAPET (TYP)
HATCH 10'x10'
KEY PLAN
RECONCILED FCN/FCR
RELEASED FOR CONSTRUCTION
REVISED & REDRAWN
BECHTEL
SAN FRANCISCO
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION
EQUIPMENT LOCATION
SERVICE & RADWASTE AREA
PLAN EL. 153'-0" & EL. 155'-3"
10855
P-0036-0
16





NOTES:

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15	SEE NOTE "A" (B-1)					

REVISION

HOPE CREEK GENERATING STATION
EQUIPMENT LOCATION
CONTROL & D/G AREA PLAN EL. 102'-0"
PLAN ELECTRICAL

PSEG NUCLEAR, L.L.C.

P-0053-0 -15

ER RESPONSIBLE FOR VERIFYING REVISION, STATUS AND CHANGES
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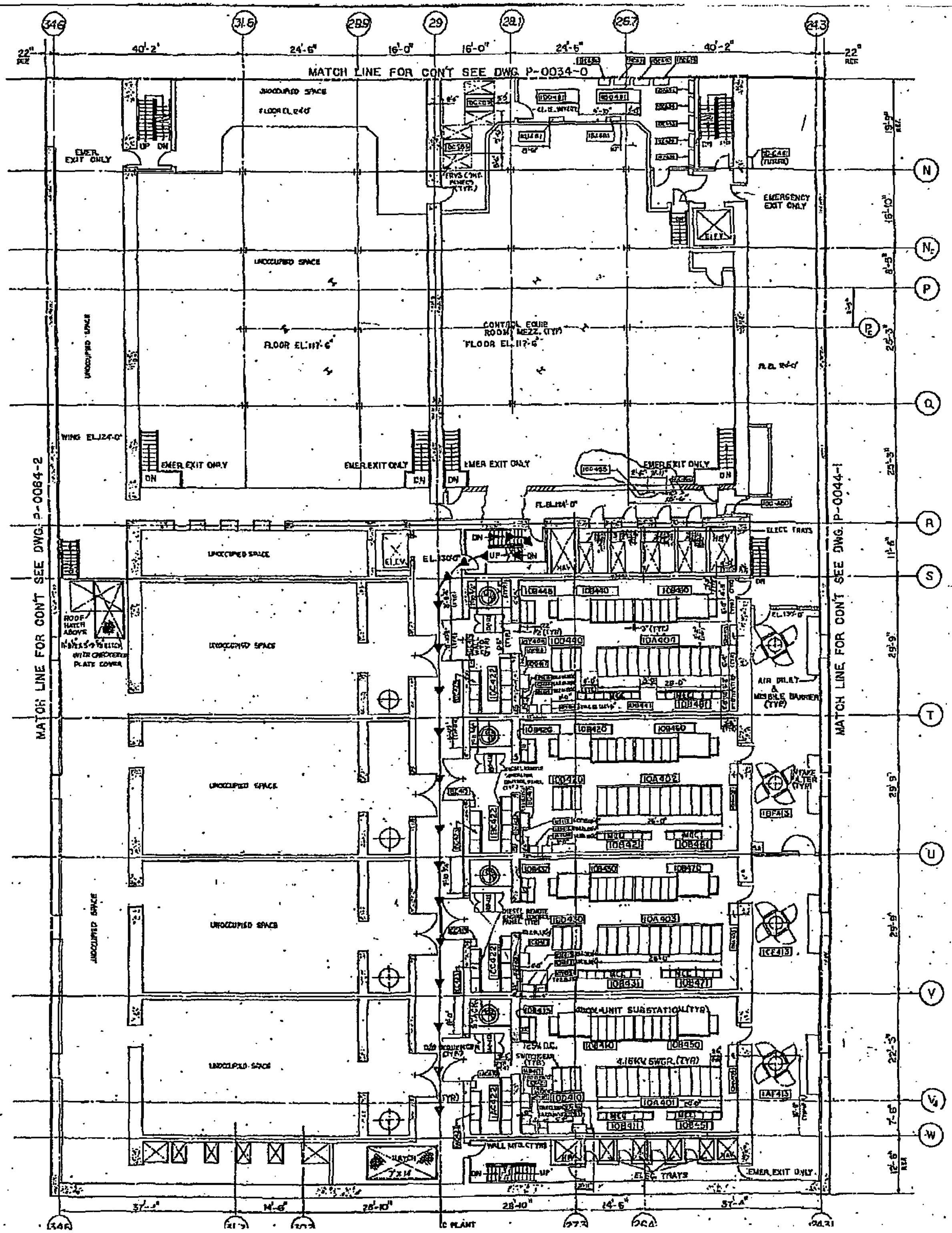

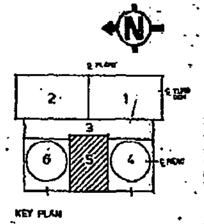

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		REVISED AS INDICATED
		ADDED [illegible] REVISED [illegible] SIZE
		ISSUED FOR CONSTRUCTION

SCALE 1/8"=1'-0"

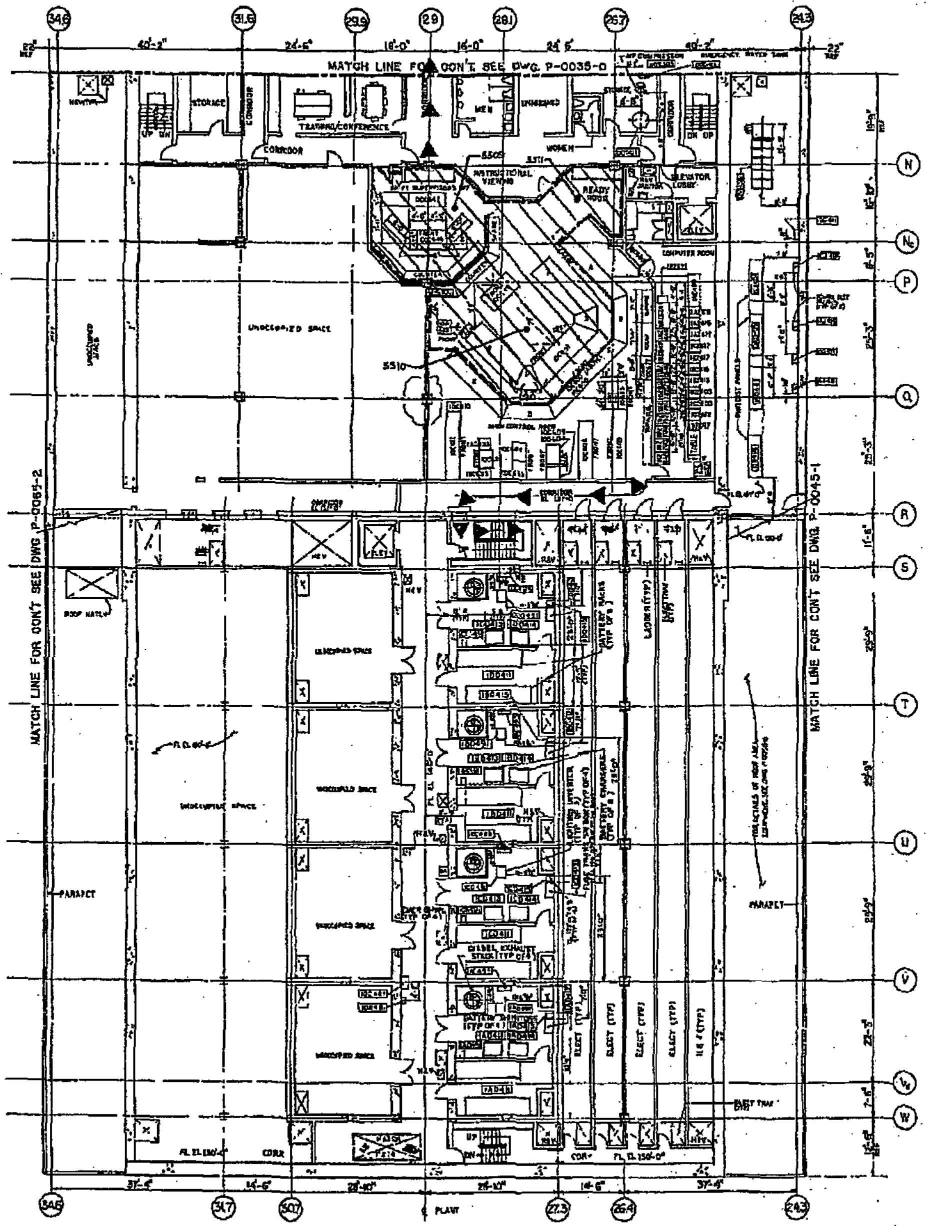
RECONCILED FCN P-15867

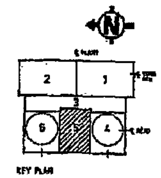
BECHTEL
SAN FRANCISCO

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

EQUIPMENT LOCATION
CONTROL & D/G. AREA
PLAN EL. 124'-0" & EL. 130'-0"

JOB No.	DRAWING No.	REV.
10855	P-0054-0	16





NOTE:

1. FOR MAIN CONTROL ROOM ARRANGEMENT AND PANEL LOCATION SEE DWGS. J-[illegible] & J-[illegible].

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NO	DESCRIPTION	DWN	PEER REV	CKD	APD	DATE
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16	SEE NOTE "A" (B-1)	DR	TG		ERC	

REVISION

HOPE CREEK GENERATING STATION
EQUIPMENT LOCATION
CONTROL & D/G. AREA
PLAN EL. 137'-0" & EL. 146'-0" & EL. 150'-0"

PSEG NUCLEAR, L.L.C.

SCALE: NONE

P-0055-0 -17