10 CFR 50.59
10 CFR 50.90
10 CFR 50.30(b)

Boston Edison
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360-5599

June 20, 1997
BECo Ltr. 2.97.066

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington DC 20555

Docket No. 50-293
License No. DPR-35

Additional Information Concerning Pilgrim Station Crediting Containment Overpressure in the Net Positive Suction Head Analysis for the Emergency Core Cooling Pumps (TAC No. M97789)

References:

1) Boston Edison Letter (BECo letter No. 97.004) to NRC dated January 20, 1997, entitled "Request for Review"
2) Boston Edison Letter (BECo letter No. 97.008) to NRC dated January 30, 1997, entitled "Significant Hazards Evaluation for Pilgrim Nuclear Power Station's Net Positive Suction Head Analyses"
3) Boston Edison Letter (BECo letter No. 97.023) to NRC dated February 27, 1997, entitled "Supplemental Submittal on Pilgrim Station NPSH Analysis"
4) Boston Edison Letter (BECo letter No. 97.042) to NRC dated April 11, 1997, entitled "Revised Request for License Amendment to Credit Containment Pressure in ECCS NPSH LOCA Analyses"
5) Safety Evaluation 2971
6) Safety Evaluation 2983
7) Calculation M662
8) GE Report GE-NE-B13-0185-11
9) NRC Request for Additional Information (RAI) dated March 13, 1997
10) NRC Request for Additional Information (RAI) dated March 31, 1997
11) NRC Request for Additional Information (RAI) dated April 17, 1997
12) Boston Edison Letter (BECo letter No. 97.035) to NRC dated May 14, 1997, entitled "Response for Request for Additional Information Concerning Pilgrim Station Crediting Containment Overpressure in the Net Positive Suction Head Analysis for the Emergency Core Cooling Pumps"

A001

By references 1 through 3, Boston Edison Company (BECo) requested NRC review and approval of a license amendment under 10 CFR 50.90 to credit containment pressure as a component of net positive suction head (NPSH) margin in the PNPS licensing basis. To aid in that review, reference 1 included submittal of NPSH and safety analyses based on the site maximum ultimate heat sink (UHS) temperature (References 5 through 8).

References 9 through 11 requested additional information (RAI) on these submittals. Boston Edison's response to the RAI questions in references 9 through 11 was provided in Attachments 1 through 4 of reference 12.

Based on recent discussions with the NRC staff regarding this license amendment, BECo is submitting supplemental information and a revised proposed limit for containment pressure credit as follows:

- Information on primary containment initial conditions is provided in Attachment 1 to this letter. This information supplements the BECo response to question 6 previously submitted in Attachment 1 to reference 12.
- The revised containment pressure request is provided in Attachment 2 to this letter and replaces the BECo response previously submitted as question 6 in Attachment 2 to reference 12. Revisions are marked with a vertical change bar in the right hand page margin.
- Attachment 3 to this letter provides information to supplement the BECo response previously provided in Attachment 3 to reference 12 regarding equipment qualification and use of the Arrhenius methodology.

Regarding the containment pressure used in NPSH analysis, BECo requests approval to credit the following levels of containment pressure when evaluating ECCS pump NPSH:

Time After Accident	Containment Pressure (psig)
0 to 1200 sec.	0
1200 to 6000 sec	1.9
6000 sec. to 30 hours	2.5

The basis for this specific request is provided in Attachment 2 to this letter.

The clarifying information provided with this letter does not change the no significant hazard consideration determination submitted by reference 2.

This letter also confirms that drywell - suppression chamber vacuum breakers are part of Pilgrim's Technical Specifications (Sections 3/4.7.A.3 and 3/4.7.A.4). The ten vacuum breakers are designed to be fully open at 0.5 psi differential between the wetwell and drywell.

This letter does not change any of the commitments previously made in reference 12.

Should you need further information on this issue, please contact P. M. Kahler at (508) 830-7939.

N L Desmond

Nancy L. Desmond
Regulatory Relations Group Manager

PJD/PMK/dmc/npshrai

Attachments:

1) Supplemental Response to BECo Response to Question 6 Previously Submitted in Attachment 1 to Reference 12. NRC Request for Additional Information (RAI) dated March 13, 1997 (Enclosure 1, questions 1 through 11, Enclosure 2, questions 1 through 10, and Enclosure 3, question 1)
2) Supplemental information to BECo Response to Question 6 Previously Submitted on Attachment 2 to Reference 12. NRC Request for Additional Information (RAI) dated March 13, 1997 (Enclosure 2, questions 11 through 16)
3) Supplemental Information to BECo Response to NRC Request for Additional Information (RAI) dated March 31, 1997 (Enclosure questions 1 through 11)

cc: Mr. Alan B. Wang, Project Manager
Project Directorate I-3
Office of Nuclear Reactor Regulation
Mail Stop: OWF 14B2
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector
Pilgrim Nuclear Power Station

Mr. Robert Hallisey
Radiation Control Program
Center for Communicable Diseases
Mass. Dept. of Public Health
305 South Street
Jamaica Plain, MA 02130

Mr. Peter LaPorte, Director
Massachusetts Energy Management Agency
400 Worcester Road
P.O. Box 1496
Framingham, MA 01701-0313
Attn: Mr. James Muckerheide

Attachment 1

Supplemental Information to BECo Response to Question 6 Previously Submitted in Attachment 1 to Reference 12

NRC Request for Additional Information (RAI) dated March 13, 1997 (Enclosure 1, questions 1 through 11, Enclosure 2, questions 1 through 10, and Enclosure 3, question 1)

Supplemental information to BECo response to Question 6:

Question 6:

Calculation M662: In Table 8 of calculation M662, in the 65 and 75 deg F columns for initial drywell relative humidity, the relative humidity is given as 80%, versus 100% in the Amendment 9 column. 80% seems less conservative from a minimum pressure perspective, since it tends to increase noncondensables in containment, resulting in a higher, not minimum, containment pressure.

Discuss why this apparently non-conservative change was made in the relative humidity.

Answer:

The change in the drywell initial relative humidity from 100% to 80% was as a result of a complete reassessment of all input assumptions as summarized in Table 8 of calculation M-662. As a result, changes were made to the SSW flow rate, the RHR and core spray pump flows and NPSH required, wetwell volumes, suction line head losses, and containment initial conditions.

The change in the drywell relative humidity (RH) was based on the conclusion that the original assumptions, although more conservative, were outside the bounds of possible conditions. The drywell is inerted with dry nitrogen for normal plant operation, and bulk temperatures are generally below 130°F. It is also required (Tech. Spec. 3.7.A.1.i) that a differential pressure of 1.17 psid minimum be maintained between the drywell and wetwell. There is a scram setpoint for high drywell pressure at 2.2 psig. Therefore, there is only a 1.03 psi increase in drywell pressure that is allowed before the reactor is automatically shutdown. If the normal condition were 150°F at 20% RH at 1.17 psig drywell pressure, then the 2.2 psig maximum would be reached simply by the change in water vapor pressure in going to 150°F at 48% RH. It is not considered a credible condition for the drywell to be at 150°F at 100% RH at 0 psig as was assumed in the original analysis. The new initial conditions are considered to be conservative since they represent an unusually high temperature and humidity at an otherwise normal drywell pressure (1.3 psig).

Changes to assumptions such as those above were made by applying the principle that design basis calculations must be conservative overall and, where appropriate, bounding for most individual parameters. However, instances should be avoided where assumptions represent improbable or impossible combinations of parameters so that the design basis does not mislead or result in unwarranted decisions or actions in response to an actual event. The current set of assumptions used in the containment heat removal analyses [Ref. 4 to this attachment] are believed to be the most appropriate and were developed from a careful review of many interrelated calculations. We believe that the analyses performed are as conservative overall as the original design basis analyses and that conservatisms are more appropriately applied to each individual parameter.

Supplemental Information

BECo has performed additional analysis to further evaluate the assumptions for containment initial conditions. To calculate a conservative initial nitrogen mass in containment, it is

necessary to establish the maximum credible temperature and humidity values for the drywell and wetwell that could exist with the reactor operating at normal full power levels. Pilgrim Technical Specifications 3.2.H give drywell temperature limits as follows:

Above elevation 40 ft.: ≤ 194°F

Equal to or Below elevation 40 ft.: ≤ 150°F

Actual drywell temperatures are monitored at 15 locations in the drywell, and the dewpoint is measured at 7 locations. The above limits represent the maximum that any one sensor may indicate per the Technical Specifications. BECo has established sets of drywell and wetwell parameters that represent typical normal conditions, maximum normal, and Technical Specification maximum values as shown on the attached tables. For each case, a bulk average temperature is calculated based on a weighting method for 8 temperature zones in the drywell. Based on the assumed pressure and humidity, the mass of nitrogen is then calculated. At greater temperature or humidity values, the mass of nitrogen decreases.

There are small fluctuations in drywell conditions during normal operation. Based on actual data, normal values have been established as a drywell average temperature of 127°F at 25% Relative Humidity (RH) with a drywell pressure of 1.3 to 1.4 psig. The wetwell normal values are 80°F at 100% RH and 0.0 psig. The resulting total mass of nitrogen is approximately 18,709 lbm. These conditions are based on a drywell dewpoint of 80°F which is typical and is also expected based on the operation of drywell fan-coil units that receive cooling water at a temperature of 75 to 80°F and effectively dehumidify the drywell to the lowest dewpoint achievable.

Maximum normal values have been established as a drywell average temperature of 135°F at 68% RH with a drywell pressure of 1.6 psig. The wetwell maximum normal values are 80°F at 100% RH and 0.2 psig. The resulting total mass of nitrogen is approximately 18,127 lbm. These values are based on the largest zones in the lower drywell being at 120°F and saturated at 100% RH such that the dewpoint is 120°F for the entire drywell volume with the upper drywell zone at 180°F (the humidity ratio, and hence dewpoint and vapor pressure, are constant for an enclosed single volume despite the temperature gradient at higher elevations). This results in the bulk average conditions of 135°F at 68% RH when the zones are mixed. The purpose of the maximum normal condition is to establish the highest temperature and dewpoint conditions that are postulated for the drywell as a result of possible venting practices, in the presence of a steam leak, prior to reaching conditions that would necessarily indicate that a steam leak exists.

FSAR Section 4.10 "Nuclear System Leakage Rate Limits" describes in detail the means provided for detection of primary system leakage. Operating procedures are designed to closely monitor primary containment parameters to ensure compliance with Technical Specifications, to detect unidentified leakage from the primary system, and to maintain containment conditions within the normal range. The following parameters are regularly monitored when reactor coolant is above 212°F:

- Drywell Air Temperatures *
- Drywell Pressure *
- Drywell Dewpoint
- Drywell Oxygen Concentration
- Drywell Air Radiation Level
- Wetwell Air Temperature *
- Wetwell Air Pressure *
- Wetwell Water Temperature *
- Wetwell Dewpoint
- Drywell Equipment & Unidentified Leakage Sump Collection
- Drywell Cooler Condensate Flow Switches

The parameters with an asterisk (*) above are displayed on chart recorders which allows easy trending of the parameters for changes. All the listed parameters except the last two are continuously monitored on the plant computer and can be plotted for trending as needed.

Any indication of primary system pressure boundary leakage requires a controlled shutdown beginning immediately upon its detection per the Technical Specifications 3.6.C. If there are indications of a primary pressure boundary leak, the reactor must be brought to at least hot shutdown condition within 12 hours and be in cold shutdown with the next 24 hours.

Fluctuations of the drywell pressure and the drywell-to-wetwell differential pressure occur during normal operations. The expected fluctuations are due to nitrogen addition to the containment from the instrument supply to components in the drywell and wetwell. Containment venting is performed from the wetwell airspace on a routine basis to lower both the drywell and wetwell pressures and to maintain the minimum differential. The wetwell is usually vented down to zero psig to maintain the drywell at approximately 1.4 psig as fixed by the submergence of the suppression pool downcomers. It is not intended that venting be used to relieve an increase in drywell pressure that would occur due to a steam leak. Therefore, the defined maximum normal condition is considered to be the highest drywell temperature and dewpoint conditions that would occur prior to commencing a shutdown due to a steam leak. The maximum normal condition is conservative because the deliberate venting of containment is required to maintain the drywell at 1.6 psig as the temperature and dewpoint increase above the normal conditions and additional venting would be needed to hold the drywell pressure at 1.6 psig as the temperatures increase further to the Tech Spec limiting values.

The maximum Tech Spec values are defined as the drywell conditions that result when the drywell is heated to the maximum temperatures of 194°F and 150°F (above and below elevation 40 ft) and steam is added to raise the drywell pressure to the scram limit of 2.2 psig. The resulting drywell average temperature is 158°F at 58% RH with a drywell pressure of 2.2 psig. The wetwell also increases to 86°F to maintain 100% RH at 0.8 psig. The resulting total

mass of nitrogen is approximately 17,811 lbm. This represents the lowest postulated initial mass of nitrogen to be considered during power operation. Although it is possible to perform additional venting and thereby increase the dewpoint to higher values, this would not be done to control containment pressure in response to a suspected leak except possibly during the controlled shutdown in order to avoid exceeding the 2.2 psig scram setpoint.

The resulting minimum postulated initial mass of approximately 17,811 lbm is greater than the design value of 17,316 lbm currently used in calculation M-662 that is based on initial drywell conditions of 150°F at 80% RH. Therefore, it is concluded that the design values used in NPSH calculations are conservative.

During the review of the older reference documents used for this evaluation, it was found that the previously referenced drywell air volume was slightly higher than a more recently calculated value (147,000 versus 138,520 ft^3). The lower volume was assumed in arriving at the 17, 811 lbm value (i.e., conservative). The future revision of the NPSH calculation M-662 will be based on the lower drywell volume such that the initial nitrogen mass will be reduced to 16,826 lbm for the same initial conditions as shown on the attached tables. The containment pressure available remains higher than the pressure requested when the reduced nitrogen mass of 16,826 lbs is used.

TABLE 1
Initial Mass of Containment Nitrogen

			Original Design Values	Current* Design Values	Future Design Values	Normal Values	Max Normal Values	Tech Spec Values	Max Tech Spec Values
P_{DW}	=	Initial Drywell pressure, psig	0	1.3	1.3	1.3	1.6	1.6	2.2
P_{WW}	=	Initial Wetwell pressure, psig	0	0	0	0	0.2	0.2	0.8
P_{VD}	=	Partial pressure of vapor in Drywell, psia	3.719	2.976	2.976	0.513	1.726	1.717	2.621
P_{VW}	=	Partial pressure of vapor in Wetwell, psia	0.5069	0.5069	0.5069	0.5069	0.5069	0.5069	0.6152
T_{SAT}	=	Drywell dewpoint temperature, °F	150	141	141	80	120	120	136
W	=	Drywell humidity ratio, lbw/lba	0.2127	0.1586	0.1586	0.0223	0.0816	0.0816	0.1351
R	=	Specific Gas Constant, ft-lbf/lbm-°R	53.3	53.3	53.3	53.3	53.3	53.3	53.3
T_{DW}	=	Average temperature of Drywell, °F	150	150	150	127	135	158	158
T_{WW}	=	Average temperature of Wetwell, °F	80	80	80	80	80	80	86
V_{DW}	=	Volume of air in Drywell, ft^3	147,000	147,000	138,520	138,520	138,520	138,520	138,520
V_{WW}	=	Volume of air in Wetwell, ft^3	120,000	124,500	124,500	124,500	124,500	124,500	124,500
f_{DW}	=	Drywell relative humidity	100%	80%	80%	25%	68%	38%	58%
f_{WW}	=	Wetwell relative humidity	100%	100%	100%	100%	100%	100%	100%
M_{DW}	=	Initial mass of N2 in Drywell, lbm	7,146	8,477	7,988	9,871	9,164	8,828	8,644
M_{WW}	=	Initial mass of N2 in Wetwell, lbm	8,519	8,838	8,838	8,838	8,963	8,963	9,167
M_T	=	Total initial mass of N2, lbm	15,665	17,316	16,826	18,709	18,127	17,791	17,811

Notes:
* The current design values are those used in Calc M-662 Rev E2.

TABLE 2
Drywell Initial Conditions

				Drywell Total Volume (ft^3) =	138,520	Normal			Max. Normal		
						Dewpoint Temp (°F) =		80	Dewpoint Temp (°F) =		120
						Humidity Ratio (lbw/lba) =		0.0223	Humidity Ratio (lbw/lba) =		0.0816
						Average Temp (°F) =		127	Average Temp (°F) =		135
TE Number TE-5050	TE Elev	Azimuth	Zone	Zone Volume Fraction	Zone Volume (ft^3)	Zone Temp (°F)	Zone Degree of Saturation	Zone Weighting Temp	Zone Temp (°F)	Zone Degree of Saturation	Zone Weighting Temp
A	87	0	II	0.0279	3,868	170	0.05	4.7	180	0.12	5.0
B	87	180	II	0.0279	3,868	170	0.05	4.7	180	0.12	5.0
C	91	50	I	0.0173	2,402	170	0.05	2.9	180	0.12	3.1
D	91	330	I	0.0173	2,402	170	0.05	2.9	180	0.12	3.1
E	60	270	III	0.0493	6,831	150	0.11	7.4	160	0.27	7.9
F	60	90	III	0.0493	6,831	150	0.11	7.4	160	0.27	7.9
G	39	270	IV	0.1326	18,368	120	0.27	15.9	140	0.53	18.6
H	39	90	IV	0.1326	18,368	120	0.27	15.9	140	0.53	18.6
J	32	0	VIII	0.0101	1,405	140	0.15	1.4	145	0.45	1.5
K	28	180	VIII	0.0101	1,405	140	0.15	1.4	145	0.45	1.5
L	22	205	V	0.1360	18,840	120	0.27	16.3	120	1.00	16.3
M	22	45	V	0.1360	18,840	120	0.27	16.3	120	1.00	16.3
N	14	270	VI	0.1074	14,880	120	0.27	12.9	120	1.00	12.9
O	15	0	VII	0.0385	5,335	100	0.52	3.9	120	1.00	4.6
P	12	125	VI	0.1074	14,880	120	0.27	12.9	120	1.00	12.9
				1.0000	138,520			127.1			135.2

TABLE 2 continued
Drywell Initial Conditions

		Drywell Total Volume (ft^3) =			138,520	Tech Spec Limit			Max. Tech Spec Limit		
						Dewpoint Temp (°F) =		120	Dewpoint Temp (°F) =		136
						Humidity Ratio (lbw/lba) =		0.0816	Humidity Ratio (lbw/lba) =		0.1351
						Average Temp (°F) =		158	Average Temp (°F) =		158
TE Number TE-5050	TE Elev	Azimuth	Zone	Zone Volume Fraction	Zone Volume (ft^3)	Zone Temp (°F)	Zone Degree of Saturation	Zone Weighting Temp	Zone Temp (°F)	Zone Degree of Saturation	Zone Weighting Temp
A	87	0	II	0.0279	3,868	194	0.06	5.4	194	0.10	5.4
B	87	180	II	0.0279	3,868	194	0.06	5.4	194	0.10	5.4
C	91	50	I	0.0173	2,402	194	0.06	3.4	194	0.10	3.4
D	91	330	I	0.0173	2,402	194	0.06	3.4	194	0.10	3.4
E	60	270	III	0.0493	6,831	194	0.06	9.6	194	0.10	9.6
F	60	90	III	0.0493	6,831	194	0.06	9.6	194	0.10	9.6
G	39	270	IV	0.1326	18,368	150	0.38	19.9	150	0.64	19.9
H	39	90	IV	0.1326	18,368	150	0.38	19.9	150	0.64	19.9
J	32	0	VIII	0.0101	1,405	150	0.38	1.5	150	0.64	1.5
K	28	180	VIII	0.0101	1,405	150	0.38	1.5	150	0.64	1.5
L	22	205	V	0.1360	18,840	150	0.38	20.4	150	0.64	20.4
M	22	45	V	0.1360	18,840	150	0.38	20.4	150	0.64	20.4
N	14	270	VI	0.1074	14,880	150	0.38	16.1	150	0.64	16.1
O	15	0	VII	0.0385	5,335	150	0.38	5.8	150	0.64	5.8
P	12	125	VI	0.1074	14,880	150	0.38	16.1	150	0.64	16.1
				1.0000	138,520			158.3			158.3

Attachment 2

Supplemental Information to BECo Response to Question 6 Previously Submitted in Attachment 2 to Reference 12.

NRC Request for Additional Information (RAI) dated March 13, 1997 (Enclosure 2, questions 11 through 16)

Supplemental information to BECo response to Question 6:

Question 6:

Justify that your use of the ANS 5.1 model for decay heat is conservative by showing that at least two standard deviations of confidence in your analyses results is provided.

Answer:

The ANSI/ANS 5.1 method for calculating decay heat requires input values for the fuel total exposure, irradiation time, enrichment, rated power, the total energy per fission Q, and the actinide production R-factor. The method determines the decay heat power after shutdown for the fission product isotopes, corrects for the effect of neutron capture in fission products, and adds the decay heat from the heavy elements (actinides). The fission product decay heat is determined for the three principal isotopes U235, U238, and Pu239 with the contribution from all other isotopes included within these three. The calculation of fission product decay heat power for each isotope $f_i(t)$ may also have an uncertainty factor applied to account for the statistical variability of the experimental data. It may be appropriate to include this added uncertainty when the decay heat power is to be used to evaluate core cooling to determine fuel peak clad temperature (PCT) for a DBA-LOCA. For core cooling, the variation of decay heat power between fuel bundles is important, and the highest potential power level on a localized basis must be determined to find the maximum possible PCT.

The considerations for the DBA-LOCA containment heat removal analysis are somewhat different than for a core cooling analysis. For the containment analysis, all decay heat parameters are expressed as normalized core average values. The reactor core is modeled as a single bulk heat source in containment, and the heat transfer considerations are all long term relative to the point of peak fuel clad temperature which occurs very early in the event. Use of the calculated fission product decay heat power directly in terms of the core average value for this type of long term analysis is consistent with the overall modeling approach which assumes uniform mixing of the fluids in the reactor vessel and throughout the containment. This is based on the assumption that statistical variations between actual bundles may affect local power levels but not the core-wide average. It is concluded that the ANSI/ANS 5.1 method provides an accurate value for the core average fission product decay heat when appropriate conservatism is included in the assumptions for fuel exposure, irradiation time, reactor operating power level, total energy per fission Q, and the G-factor to account for neutron capture.

The decay heat contribution from the actinides (U239 and Np239) is calculated per ANSI/ANS 5.1 based on given values for the average decay energy per atom (MeV), the decay constant, and the assumed value for the actinide production factor (R-factor). The R-factor increases with fuel exposure, and the value used is conservative based on the exposures assumed. Therefore, no additional uncertainty is added to the decay heat from actinide decay.

The statistical certainty of the overall total decay heat profile is difficult to quantify since each input parameter and assumption involves a different level of conservatism and uncertainty. The only parameter for which a strictly statistical approach is used is the fission product decay power for the individual isotopes U235, U238, and Pu239 where the standard deviation is

given. When all input parameters and assumptions are chosen with appropriate conservatism for a long term containment heat removal analysis, the total decay heat determined is sufficiently conservative without specifying two standard deviations of confidence for the isotopic fission product decay terms.

Also, since decay heat is only one parameter in the containment heat removal analysis, it is necessary to consider the overall method to be used. The modeling of the reactor and containment and the other sensible heat addition sources are also conservatively specified. The heat removal part of the analysis consists of specifications for pump and heat exchanger performance that are known to significantly underestimate the abilities of these systems to transfer heat from the suppression pool to the ultimate heat sink (UHS). The peak suppression pool temperature is dependent on a balance between the heat additions and cooling capacities. While the design margins for the heat removal systems are quantifiable and controllable, an increase in decay heat sources due solely to added uncertainty decreases the available margin arbitrarily.

It is also noteworthy to compare the uses of a core cooling analysis to determine PCT versus a containment heat removal analysis. The core cooling analysis must verify that the engineered safety systems, upon automatic initiation, will maintain adequate PCT during and after the core reflooding. The containment analysis considers long term heat removal and is based on many assumptions regarding operator actions as well as heat transfer equipment performance. As such, it is important that this analysis predict the potential consequences of the DBA-LOCA with an appropriate, but not arbitrary, level of conservatism.

For Pilgrim, a benchmarking approach was used to evaluate the current methods relative to the original FSAR analysis for the DBA-LOCA. As described earlier (question #2), the current analysis includes significant changes from the original FSAR analysis. However, if all assumptions, including those for decay heat, used in the original FSAR DBA-LOCA analysis were maintained equal, and only the UHS temperature was changed 10°F from 65°F to 75°F, then it would be expected that the peak suppression pool temperature would increase from 166°F by 10°F or less to a new peak of approximately 176°F. The new GE DBA-LOCA analysis, using a 75°F UHS, resulted in a peak suppression pool temperature of 178°F.

Other factors that contribute to the overall conservatism of this analysis are tabulated in Table 1 located at the end of this attachment. These factors contribute to the overall conservatism by increasing the heat addition to the containment or decreasing the heat removal achieved by containment cooling and other heat loss mechanisms.

Considering all these factors, it was BECo's conclusion that the new analysis that used a nominal decay heat is conservative in comparison to the original FSAR analysis.

An analysis of the DBA-LOCA with 2σ uncertainty added to the decay heat will result in a more rapid containment heatup and higher peak suppression pool temperature than resulted from the analysis discussed in SE 2983 which was based on the nominal decay heat described above. The peak suppression pool temperature resulting from the DBA-LOCA with 2σ uncertainty added to the decay heat will be less than 185°F which is 7°F higher than the 178°F

peak pool temperature resulting from analysis that used a nominal decay heat without uncertainty added. This estimate of 185°F was calculated by adding to the suppression pool temperature for the nominal decay heat case (i.e., 178°F) the temperature change resulting from the additional decay heat energy from the 2σ uncertainty adder. That is, the additional decay heat energy due to the 2σ adder was integrated over finite time increments and then added to the suppression pool water to increase its temperature. No accounting was made of the higher heat removal rates (due to the higher temperature across the RHR heat exchanger) that will occur between the time of containment cooling initiation and the time that the peak suppression pool temperature occurs at approximately 20,000 seconds. No accounting is made of the portion of the additional energy that will be stored elsewhere inside the containment, to be removed later in the cooldown. All of the additional energy from the 2σ uncertainty adder between 0 to 20,000 seconds is assumed to be present in the suppression pool at 20,000 seconds, thereby, raising the suppression pool temperature to a conservative estimated maximum of 185°F. Figure 1 illustrates the pool temperature response calculated using the method described above.

The estimated suppression pool temperature of 185°F (based on a UHS temperature of 75°F and decay heat ANS 5.1-1979 + 2σ) is 19 degrees higher than the original licensing basis value of 166°F (based on a UHS temperature of 65°F). This peak suppression pool temperature of 185°F is well within the containment design temperature limit of 281°F and is a very conservative estimate considering that the ultimate heat sink temperature is increased only ten degree's from the original analysis.

A suppression pool temperature of 185°F will require containment overpressure to meet ECCS NPSH requirements with a clean strainer and/or the small additional head loss resulting from insulation debris. BECo installed large capacity stacked disc pump suction strainers for both the core spray and RHR pumps during the last refueling outage (RFO # 11). The current licensing basis debris-related head loss resulting from fibrous insulation is negligible because of the large surface area of the new strainers. Furthermore, BECo cleaned accumulated debris consisting of sludge and corrosion particles from the suppression pool during RFO # 11. Each of these measures, the new stacked disc strainers, and suppression pool cleaning substantially reduce and limit the potential degradation of ECCS pump suction conditions.

With regard to NPSH, the following information provides the basis for BECo's request for approval to credit the following amounts of containment pressure when evaluating ECCS pumps NPSH:

Time After Accident	Containment Pressure (psig)
0 to 1200 sec.	0
1200 to 6000 sec	1.9
6000 sec. to 30 hours	2.5

The requested values of containment pressure take into consideration a peak suppression pool temperature of 185°F, the current licensing basis debris volume and head loss across the new stacked disc strainers. The requested values of containment pressure will ensure that NPSH requirements are met for all postulated single failures including a LPCI loop select logic failure that results in four LPCI pumps injecting into the broken recirculation loop.

NPSHA is defined by the following terms:

Eq. 1
$$NPSHA = (Pc - Pvp)\frac{\left(144\frac{in^2}{ft^2}\right)}{\rho} + H_z - H_{sl} - Hdebris$$

Where:

Hz	*Elevation of suppression pool water surface above the pump inlet, ft*
Hsl	*Suction line losses, ft*
Hdebris	*Head loss from debris, ft*
NPSHA	*Net positive suction head available, feet*
NPSHR	*Net positive suction head required, feet*
Pc	*Pressure of primary containment, psia*
Pc Req'd	*Pressure of primary containment required to provide NPSHR, psia*
Pvp	*Vapor pressure at pool temperature, psia*
ρ	*Density of water in pool, lb/ft*^{3}

The containment pressure required to provide adequate NPSH is derived using Equation 1 by letting NPSHA equal NPSHR and solving for the containment pressure *Pc*. When NPSHA equals NPSHR, the containment pressure is by definition equal to the required containment pressure *Pc Req'd*.

Eq. 2
$$Pc\ Req'd = Pvp + (NPSHR - Hz + Hsl + Hdebris)\frac{\rho}{\left(144\frac{in^2}{ft^2}\right)}$$

The following input parameters are used in Eq. 2 to calculate the pressure required to provide adequate NPSH of 1.62 psig which is based on the core spray pump(s) NPSH requirement:

Input Parameter	RHR Pump	Core Spray Pump
Hz (ft)	12.5	12.5
Hsl (ft)	2.62	2.38
Hdebris (ft)	0.01	0.01
NPSHR (ft)	23	29
Pvp (psia) @ 185°F	8.386	8.386
ρ *of pumped fluid (lb/ft3) @ 185°F*	60.456	60.456
Pc Req'd for pump with maximum debris on suction strainer (psig)	-0.8 (13.896 psia)	1.62

Based on the above equations and calculation, a minimum of 1.62 psig is required to provide adequate NPSH at the estimated peak suppression pool temperature of 185°F.

In the additional question provided in (Enclosure 3 of the March 13, 1997, NRC RAI), the NRC questioned the use and purpose of Equation 10 in BECo calculation M662. Attachment 1

contains a response to this question stating that the purpose of this dP_{MAX} criteria is to provide a finite, detectable change from the normal suction pressure drop that is sufficient in magnitude to be measurable by test gages. It is required that the dP_{MAX} criteria have such a basis since exceeding the limits will result in the need to take immediate action to either shut down from power operation to inspect the strainers or otherwise remedy the situation. BECo proposes to add two (2) feet of additional head loss to the clean strainer suction line loss term (*Hsl*) to account for instrument reading variations during monthly IST test measurements of the suction line loss.

The following input parameters are used in Eq. 2 to calculate the pressure required to provide adequate NPSH of 2.46 psig which is based on the core spray pump(s) NPSH requirement. Note that *Hsl* is increased by two feet, and all other input values are unchanged from the previous calculation.

Input Parameter	RHR Pump	Core Spray Pump
Hz (ft)	12.5	12.5
Hsl (ft)	4.62	4.38
Hdebris (ft)	0.01	0.01
NPSHR (ft)	23	29
Pvp (psia) @ 185°F	8.386	8.386
ρ of pumped fluid (lb/ft3) @ 185°F	60.456	60.456
Pc Req'd for pump with maximum debris on suction strainer (psig)	0.04	2.46 Rounded to 2.5 psig

BECo previously requested that the NRC approve the analysis described in FSAR section 14.5.3.1.3 and illustrated on FSAR Figures 14.5-9 through 10, Figure 14.5-13, and Figures 14.5-18 through 19. The analysis described in the FSAR represents a lower bound estimate of the containment pressure available following a DBA-LOCA.

Figure 2 in this response, compares the requested values of containment pressure as a function of time against the containment pressure available assuming 5% leakage of noncondensible gas as illustrated on FSAR Figure 14.5-18. Also included on this figure is pressure required for the limiting ECCS pump(s) which is the core spray pump. However, the pressure required curve for the core spray pump is slightly higher than that shown on FSAR Figure 14.5-18 because the suction line loss was increased by 2 feet to accommodate IST measurement variations, and the pressure required to compensate for debris related head loss of .01 feet was included. As the figure illustrates, the pressure requested is less than that which will be available and greater than that which will be required.

Also included on Figure 2 is the estimated containment pressure available based on the ANS 5.1-1979 decay heat plus a 2 sigma uncertainty adder. The peak pressure available corresponds to the peak suppression pool temperature discussed previously in this response of 185°F. The estimated peak pressure required of 2.46 psig also corresponds with the estimated peak suppression pool temperature of 185°F.

Based on the above NPSH evaluation, BECo requests approval to credit the following amounts of containment overpressure when evaluating ECCS pump NPSH for the limiting DBA-LOCA event:

Time After Accident	Containment Pressure (psig)
0 to 1200 sec.	0
1200 to 6000 sec	1.9
6000 sec. to 30 hours	2.5

REFERENCES

1. GE Report GE-NE-T23-00732-01 "Containment Heat Removal Analysis", March 1996, SUDDS/RF # 96-05.
2. GE Specification 23A6938 Rev. 1 "Decay Heat Requirements", June 1992.
3. ANSI/ANS 5.1-1979 "Decay Heat Power in Light Water Reactors", August 1979.
4. GE NEDO-23729 "Nuclear Basis for ECCS (Appendix K) Calculations", November 1977.
5. GE NEDC-23785P "The GESTR-LOCA and SAFER Models for the Evaluation of Loss-of-Coolant Accident", October 1984.

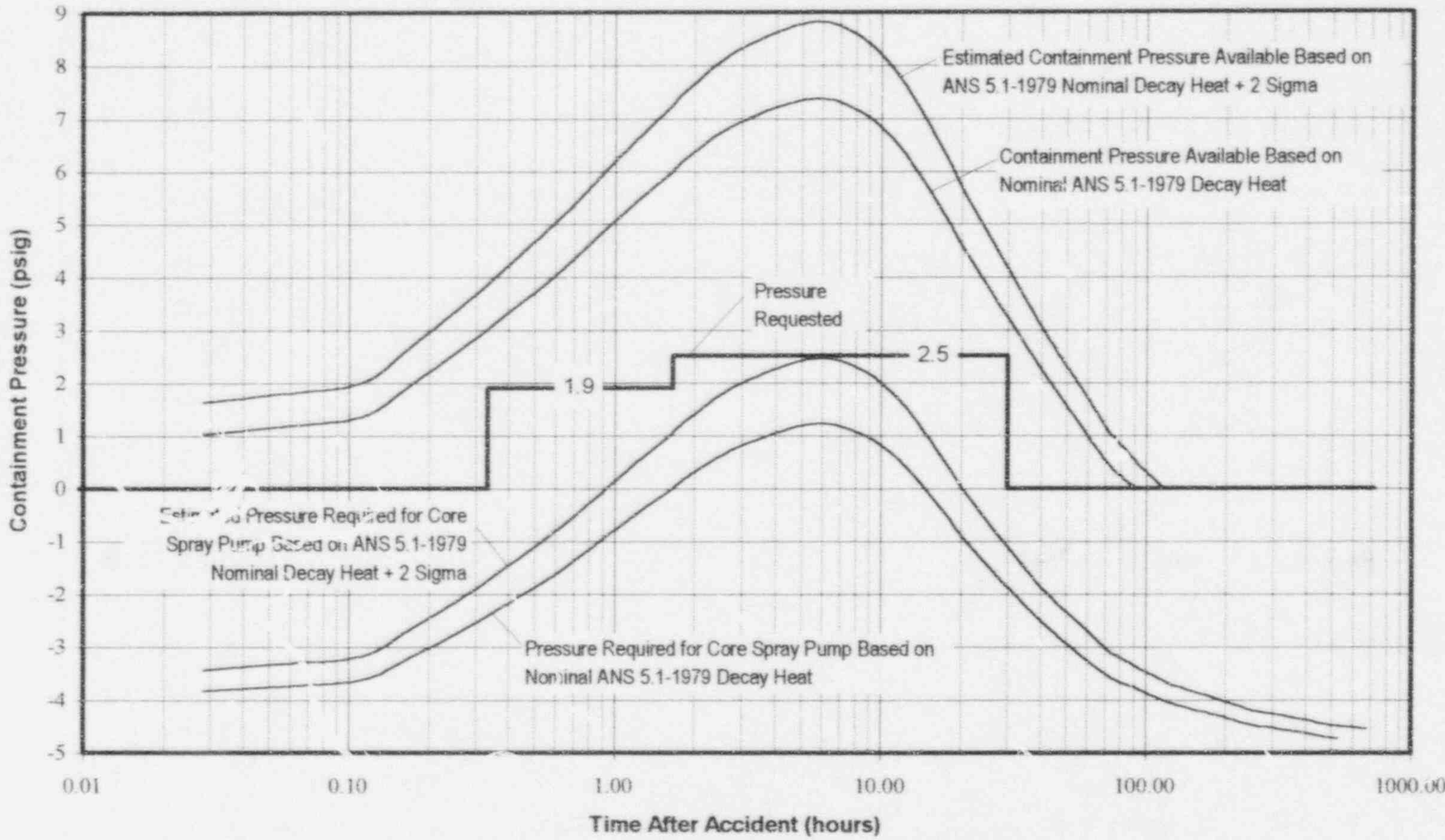
Table 1

Parameter	Original Licensing Basis DBA-LOCA	DBA-LOCA Analysis SE2983	Two-Sigma DBA-LOCA Analysis
Decay Heat	Undocumented in licensing basis. May-Witt used for benchmark analysis.	ANS 5.1-1979 without uncertainty	ANS 5.1-1979 with two sigma uncertainty
Initial power level	100%	102%	102%
Feedwater Input	Minimized to limit vessel depressurization from colder feedwater which will lower the peak short-term containment pressure.	This is a long-term heat up analysis, so feedwater addition was continued while the feedwater enthalpy contributes to suppression pool heatup. This treatment maximizes suppression pool temperature and the secondary peak pressure.	This is a long-term heat up analysis, so feedwater addition was continued while the feedwater enthalpy contributes to suppression pool heatup. This treatment maximizes suppression pool temperature and the secondary peak pressure.
Reactor Vessel Conditions	Saturated	Saturated	Saturated
Torus water level	Technical Specification minimum	Technical Specification minimum	Technical Specification minimum
ECCS pump heat addition	None	Rated horsepower is added when pump(s) are operating	Rated horsepower is added when pump(s) are operating
Heat loss from containment	Containment is assumed to be perfectly insulated, heat removed only via the RHR heat exchanger	Containment is assumed to be perfectly insulated, heat removed only via the RHR heat exchanger	Containment is assumed to be perfectly insulated, heat removed only via the RHR heat exchanger
Internal heat sinks	Not included	Not included	Not included
Heat exchanger Performance	Based on original heat exchanger ratings at rated flow conditions and design fouling.	Based on heat exchanger performance at system minimum flow rates and maximum allowable fouling/tube plugging that are design basis for in-service performance testing for pumps and heat exchangers.	Based on heat exchanger performance at system minimum flow rates and maximum allowable fouling/tube plugging that are design basis for in-service performance testing for pumps and heat exchangers.
Containment Cooling Initiation Time and Flows	at 600 seconds with 5000 gpm through the RHR heat exchanger	at 600 seconds with 3430 gpm, switching to 5100 gpm at 2 hours.	at 600 seconds with 3430 gpm, switching to 5100 gpm at 2 hours.
Service Water Inlet Temperature	Constant 65°F	Constant 75°F for 30 days. No accounting for temperature variations from the diurnal cycle and tidal effects.	Constant 75°F for 30 days. No accounting for temperature variations from diurnal cycle and tidal effects.

Table 1 Continued...
Key Results

Parameter	Original Licensing Basis DBA-LOCA	DBA-LOCA Analysis SE2983	Two-Sigma DBA-LOCA Analysis
Peak Suppression Pool Temperature	166°F	178°F	Estimated at 185°F
Secondary Peak Containment Pressure	8.0 psig	9.7 psig	Estimated at 10.9 psig
Time of Occurrence for Peak Suppression Pool Temperature and Secondary Pressure	5.5 hours	5.5 hours	Estimated at 5.5 hours. Should occur earlier with a higher decay heat source.

Figure 2 - Comparison of Requested Pressure to Containment Pressure Available and Containment Pressure Required



Note: Curves are based on a 75°F ultimate heat sink temperature, and pressure required curves include a 2 foot additional suction line loss to accommodate IST measurement variation, and a 0.01 foot debris related head loss.

Attachment 3

Supplemental Information to BECo Response to NRC Request for Additional Information (RAI) dated March 31, 1997
(Enclosure questions 1 through 11)

SUPPLEMENTAL RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)
DATED MARCH 31, 1997
(ENCLOSURE QUESTIONS 1 THROUGH 11)

The following information is in response to additional questions concerning the application of the Arrhenius Methodology.

Concern: The applicability of the Arrhenius Methodology

Pilgrim Station and our material consultant have concluded that using the Arrhenius Methodology when extrapolating operating times is a valid approach due to the temperature rating of the materials used. This review was conducted for all material located in the drywell at PNPS. Based upon published technical information, fundamental considerations of polymer science and chemistry, direct experience with the materials, and our consultants experience with polymeric materials and the aging characteristics of these materials, it was concluded that the material properties would not change such that the use of the Arrhenius Methodology would be invalid. For nuclear plant containment applications, virtually all elastomers and most thermoplastics are crosslinked. The crosslinking process not only eliminates polymer melting but increases its heat aging resistance, chemical resistance, and physical durability under a wide range of conditions. Accordingly, the materials can be used at high temperatures without undergoing chemical changes; therefore, the Arrhenius Methodology is acceptable to use after the equipment has been subjected to LOCA temperatures.

Pilgrim Station utilizes the Arrhenius Methodology for those cases where the test profile does not envelop the accident profile for the required duration (i.e., 30 days + 10% margin = 33 days). In many instances, the Pilgrim Station required operating time is less than 30 days. The Pilgrim Station EQ Design Guide, as well as IEEE 323-1974, requires "an equipment operating time of +10% of the period of time the equipment is required to be operational following the start of the DBE" for equipment required to satisfy 10CFR50.49. In addition, most equipment is tested to more than one transient. PNPS does not use the first transient when utilizing the Arrhenius methodology for extending operating time. For these considerations, all in-containment equipment required to satisfy 10CFR50.49 has a minimum margin of approximately a factor of two greater than required (see below for further discussion). Some equipment has margins that are many factors above the expectation.

Concern: Utilization of the Arrhenius Methodology at Greater than One Hour Past the Transient Peak Temperature.

Pilgrim Station, typical of the nuclear industry, utilizes the Arrhenius Methodology as a tool within their EQ program. Pilgrim Station has reviewed all in-containment EQML equipment utilizing the Arrhenius Methodology[where required] starting at times greater than one hour past the transient peak temperature. The assessment was focused on determining whether all in-containment EQ equipment would be qualified with a two(2) times the 10% margin required by IEEE-323-1974. All equipment satisfies both of these criteria except for the internal Rockbestos wire associated with Limitorque limit switches. These limit switches are associated with the R.G.1.97 program and are used for position indication only. Further, this equipment is qualified to the DOR Guidelines [10CFR50.49] which does not require the inclusion of 10% margin. As part of the EQ process, Pilgrim Station has qualified this equipment and included 10% margin for further conservatism.

Although Pilgrim Station can meet the 2 times 10% margin criteria with the lone exception noted, we will continue to comply with the 10% margin criteria established in IEEE 323-1974 for equipment qualification under 10CFR50.49. This criteria is used throughout the nuclear industry for EQ programs that comply with 10CFR50.49 and is consistent with the PNPS licensing basis.