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

MAY 2 0 2005

PSEG
Nuclear LLC

LR-N05-0213
LCR H04-01

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
ARTS/MELLLA IMPLEMENTATION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Reference: LR-N04-0062, "Request for License Amendment: ARTS/MELLLA Implementation," dated June 7, 2004.

By the referenced letter, PSEG Nuclear LLC (PSEG) requested a revision to the Technical Specifications (TS) for the Hope Creek Generating Station to reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). In the referenced letter, PSEG also proposed to make changes in the methods used to evaluate annulus pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB).

In a communication from Mr. D. Collins on February 16, 2005, and in a telephone conference on April 1, 2005, the NRC requested additional information concerning the proposed change. The information requested by the Containment and Accident Dose Assessment Section is provided in Attachment 1 to this letter. In accordance with 10 CFR 50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

Attachment 1 contains proprietary information as defined by 10 CFR 2.390. General Electric Company (GE), as the owner of the proprietary information, has executed the affidavit included in Attachment 1, which identifies that the attached proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to PSEG in a GE transmittal that is referenced by the affidavit. The proprietary information has been faithfully reproduced in the attached RAI responses

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such that the affidavit remains applicable. GE requests that the proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17. A non-proprietary version of the RAI responses also is provided in Attachment 2. A copy of NEDO-24548, "Technical Description - Annulus Pressurization Load Adequacy Evaluation," is provided in Attachment 3.

PSEG has determined that the information contained in this letter and attachment does not alter the conclusions reached in the 10 CFR 50.92 no significant hazards analysis previously submitted.

If you have any questions or require additional information, please contact Mr. Paul Duke at (856) 339-1466.

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

Executed on May 20, 2005
(date)

George P. Barnes
George P. Barnes
Site Vice President
Hope Creek Generating Station

Attachments (3)

C: Mr. S. Collins, Administrator – Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. D. Collins, Project Manager - Salem & Hope Creek
U. S. Nuclear Regulatory Commission
Mail Stop 08C2
Washington, DC 20555

USNRC Senior Resident Inspector – Hope Creek (X24)

Mr. K. Tosch, Manager IV (without Attachment 1)
Bureau of Nuclear Engineering
PO Box 415
Trenton, New Jersey 08625

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
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ARTS/MELLLA IMPLEMENTATION

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In a communication from Mr. D. Collins on February 16, 2005, and in a telephone conference on April 1, 2005, the NRC requested additional information concerning the proposed change. The information requested by the Containment and Accident Dose Assessment Section is provided below.

1. Containment and Accident Dose Assessment Section

   a) The current licensed thermal power (CLTP) is 3339 MWt. The original licensed thermal power (OLTP) was 3293 MWt. The referenced updated final safety analysis report analyses were performed at 102% of the OLTP, which suggests that the containment studies would be equivalent to 100.6% of the CLTP. Clarify this apparent discrepancy or provide justification for reducing the conservatism in the containment performance studies from the previous value of 102% of the nominal operating power to 100.6% of the nominal operating power.

      PSEG Response:
      The current licensed thermal power level (3339 MWt) is based on reduced uncertainty in core thermal power measurement achieved with the Crossflow ultrasonic flow measurement system. The power measurement uncertainty is less than 0.6%. The NRC approved the increase in the authorized maximum power level in License Amendment No. 131 (TAC No. MB0644).

   b) In your submittal, you reference General Electric Co. (GE) Nuclear Energy, "Technical Description - Annulus Pressurization Load Adequacy Evaluation," NEDO-24548, January 1979. Please provide a copy of this report to assist the staff in its review of the proposed [change.]

      PSEG Response:
      Attachment 3 contains a copy of NEDO-24548.

c) Provide a discussion of the differences in the calculation of the subcompartment loads previously used for the reactor asymmetric loads evaluation (the COPDA code) to the proposed revised method using the COMPARE code. Include, for example, the treatment of heat structures, the treatment of the water-steam-air mixture for mass and heat transfer, and the treatment of the mass and energy as they enter the break control volume for mass and energy partitioning between the fluid and vapor states. Provide comparison graphs of the limiting control volume and differential pressures, over the time period of interest, for both methods.

PSEG Response:
The design basis analysis used the proprietary code COPDA. For this license amendment, COMPARE Mod 1 (LA-7199-MS) was used.

The COMPARE Mod 1 analysis used the equilibrium mixture of air-steam-water option for each node and the junction inertial flow option. The heat transfer option was not used. COMPARE assumes a homogeneous steam-water mixture in all nodes. Therefore, the COMPARE Mod 1 basically duplicates COMPARE (LA-NUREG-6488-MS) response for subcompartment analysis.

The original licensing basis analysis in HCGS UFSAR Appendix 6B was performed using COPDA assuming a 50% - 50% flow split between the annulus and the drywell for the recirculation suction line break. A subsequent COPDA analysis using a flow split of 25% - 75% consistent with the current HCGS diverter was performed and documented that the original licensing basis analysis remained bounding.

COMPARE and COPDA pressure and net force comparisons were made for the 25% - 75% flow split case. COMPARE calculated pressure time histories that resulted in a total vessel force within approximately 7 percent of the maximum force predicted using COPDA's pressure time histories. In general, the COPDA analysis typically resulted in higher nodal pressures and differential pressures in the vicinity of the blowdown nodes. In the COMPARE analyses, the break mass and energy was more rapidly distributed resulting in higher nodal pressures away from the blowdown site for the time period from the break opening through the time of maximum resultant force. However, the resultant forces using the pressure results predicted by the COPDA and COMPARE codes were similar through the time of maximum resultant force. These forces were calculated in the same manner as described in Appendix 6B of the UFSAR.

Comparisons of significant (in terms of resultant force contribution) control volume and differential pressures were made for COMPARE and COPDA. UFSAR Figure 6B-3a shows the nodal layout and is provided in this attachment as Figure 1c-8. The recirculation outlet is located at the boundary between nodes 13 and 24. One half of the recirculation line break annulus blowdown goes into nodes 13 and 24, respectively. Node 14 is on the same elevation adjacent to node 13 going away from the blowdown region. Node 1 is immediately below node 13 extending to the vessel skirt; nodes 25 and 37 are immediately above (in elevation) node 13. Nodes 19, 20, 7, 31 and 43 are diametrically on the opposite side of the vessel from these nodes and result in larger differential pressures relative to the break location and elevation.

Figures 1c-2 through 1c-7 show comparisons of significant pressure nodes and differential pressures predicted by COMPARE and COPDA through the time of maximum resultant force for the 25%-75% match case. Pressures for nodes 24 and 18 are similar to nodes 13 and 19, etc. due to symmetry (see UFSAR Figure 6B-3a), and are not shown in the plots.

Figure 1c-1 shows resultant force time histories similar to UFSAR Figure 6B-8. These comparisons show the HCGS design basis (50% - 50% flow split) and the COPDA and COMPARE match cases with 25% - 75% flow splits. Also, shown for comparison is the reactor vessel resultant force for MELLLA LAMB M/E with a 25%-75% split.

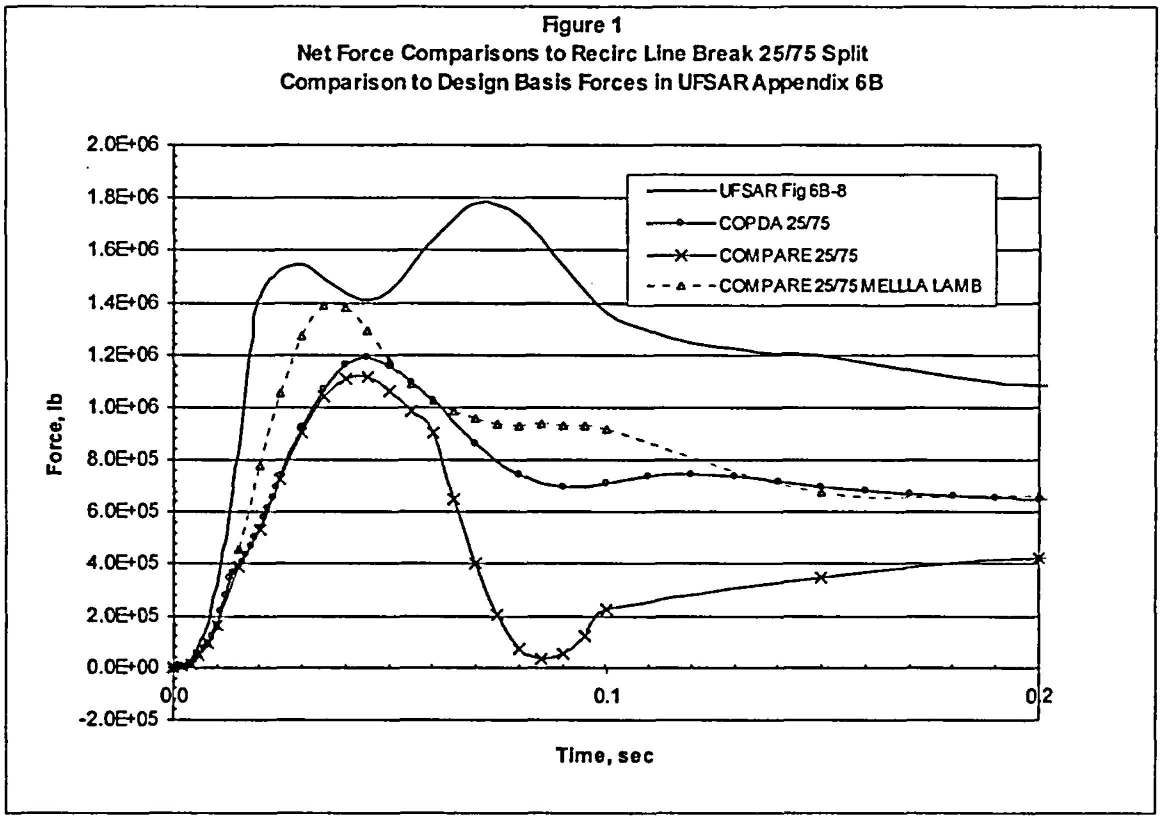
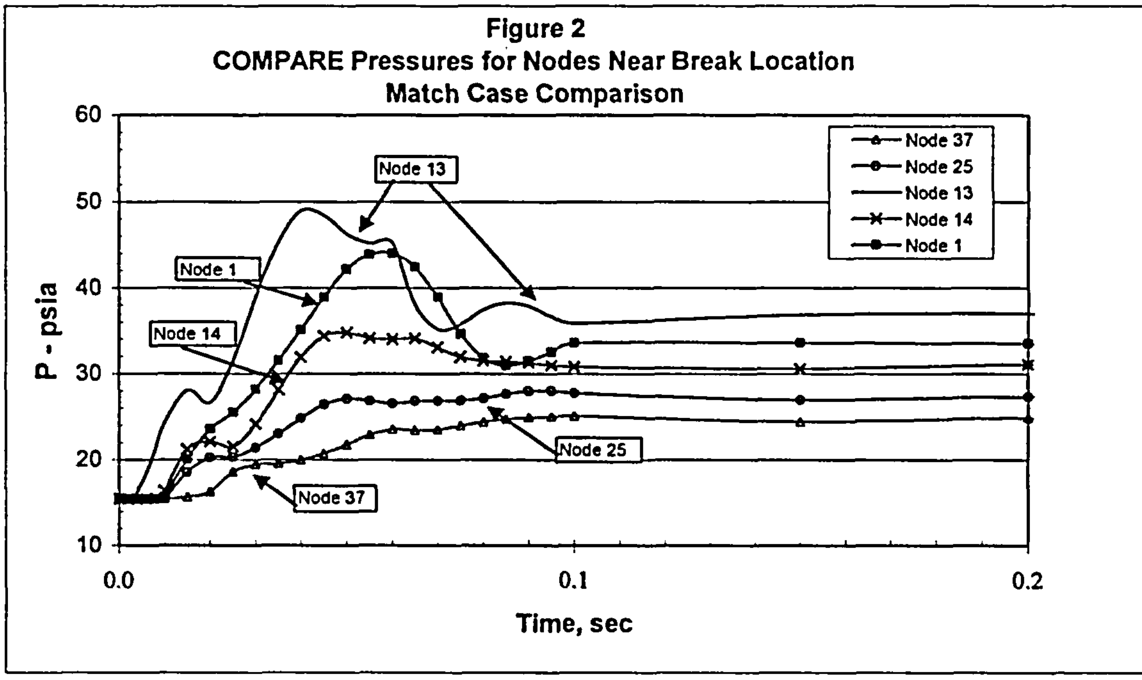
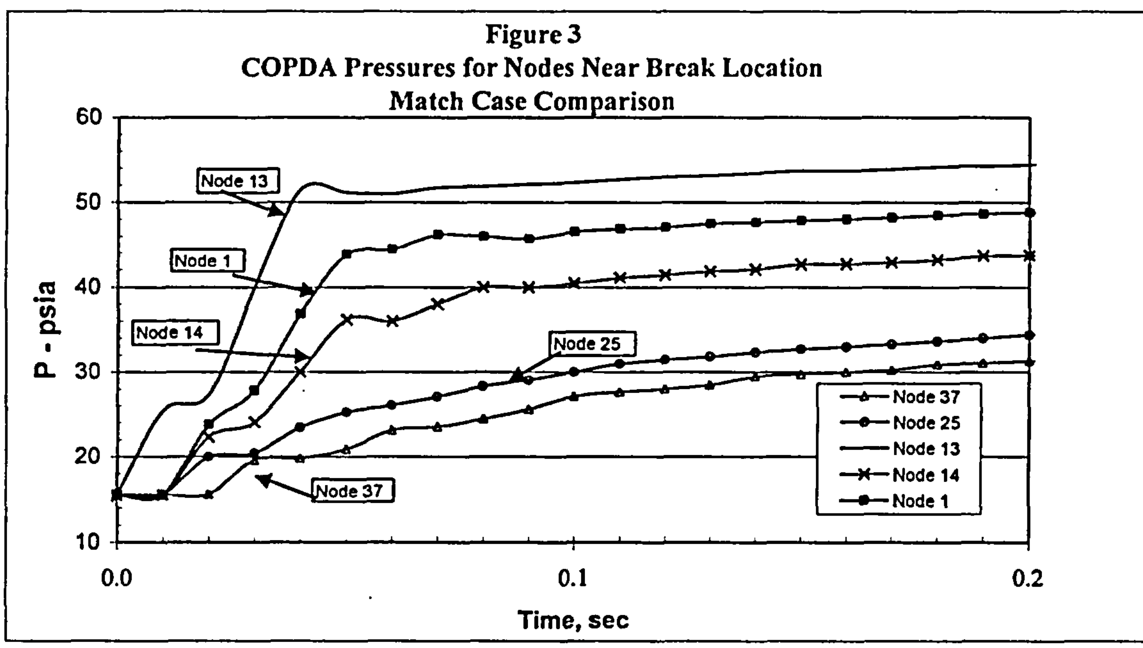


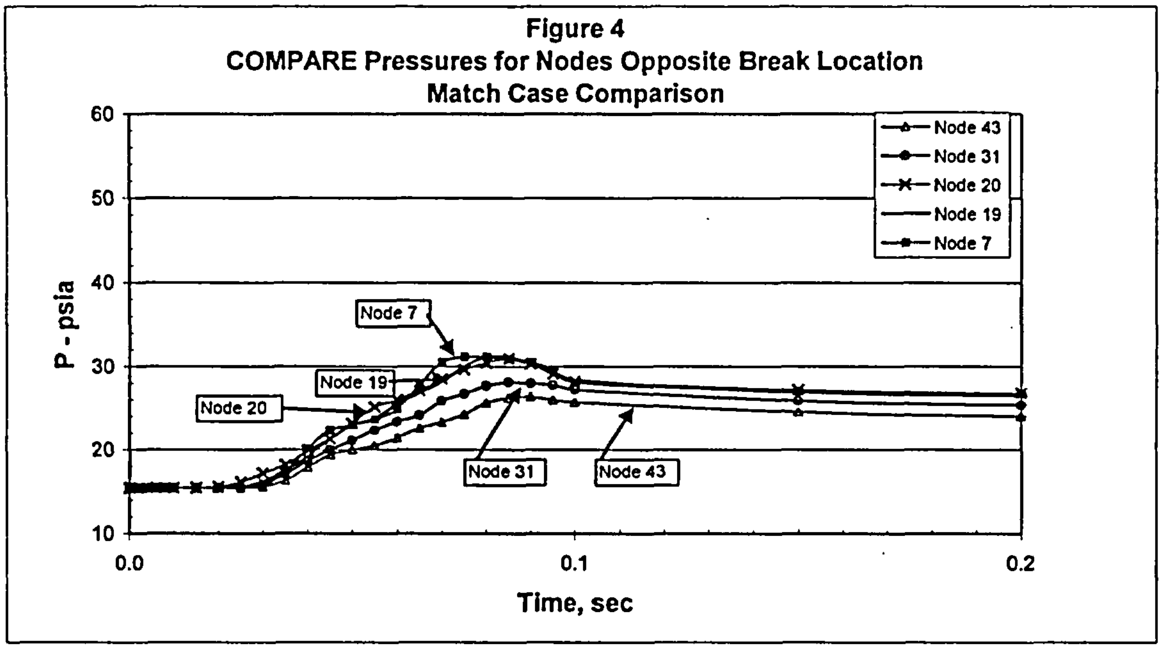
Figure 1
Net Force Comparisons to Recirc Line Break 25/75 Split
Comparison to Design Basis Forces in UFSAR Appendix 6B
UFSAR Fig 6B-8
COPDA 25/75
COMPARE 25/75
COMPARE 25/75 MELLLA LAMB
2.0E+06
1.8E+06
1.6E+06
1.4E+06
1.2E+06
1.0E+06
8.0E+05
6.0E+05
4.0E+05
2.0E+05
0.0E+00
-2.0E+05
Force, lb
0.0
0.1
0.2
Time, sec

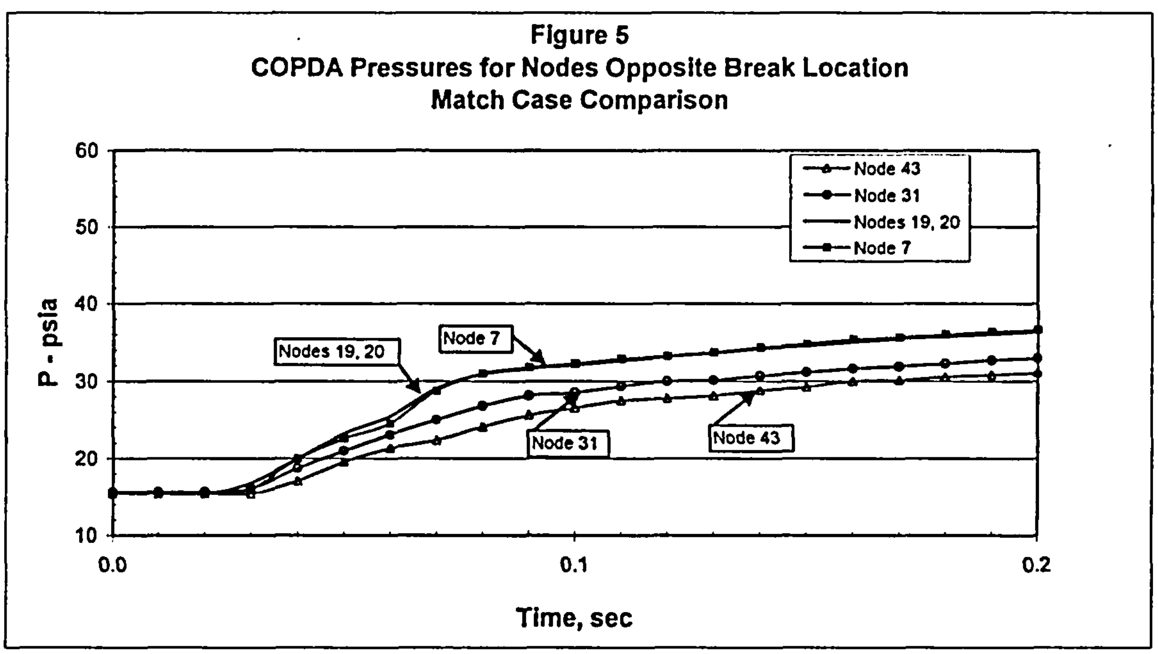
Figure 1c-1



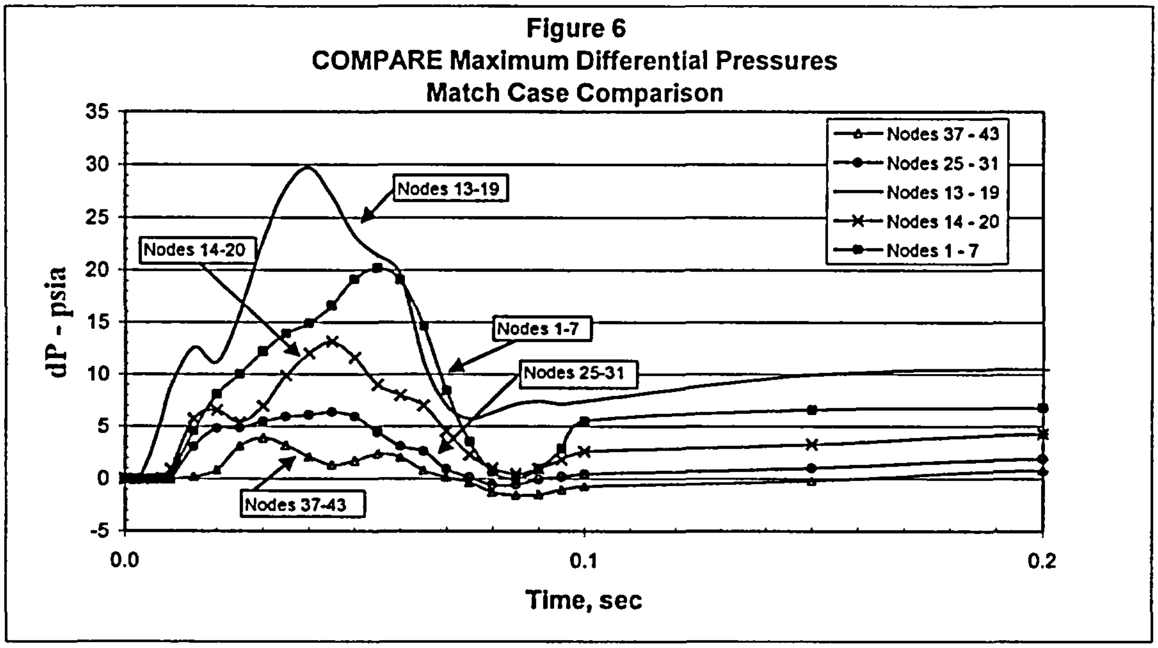


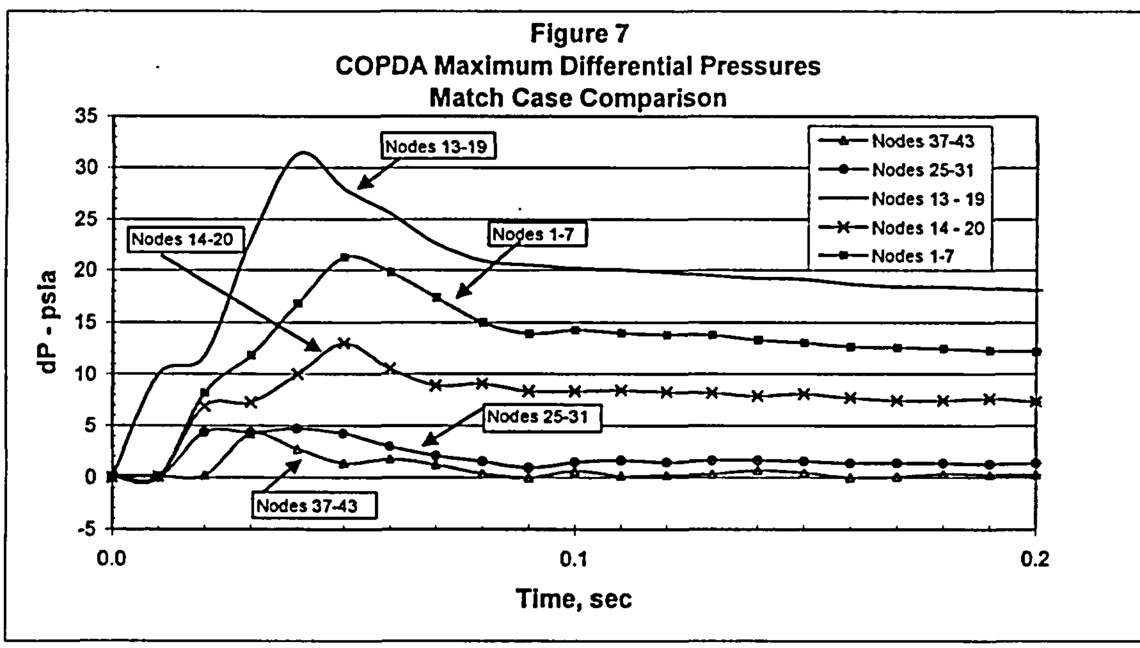
Figures 1c-2 and 1c-3





Figures 1c-4 and 1c-5





Figures 1c-6 and 1c-7

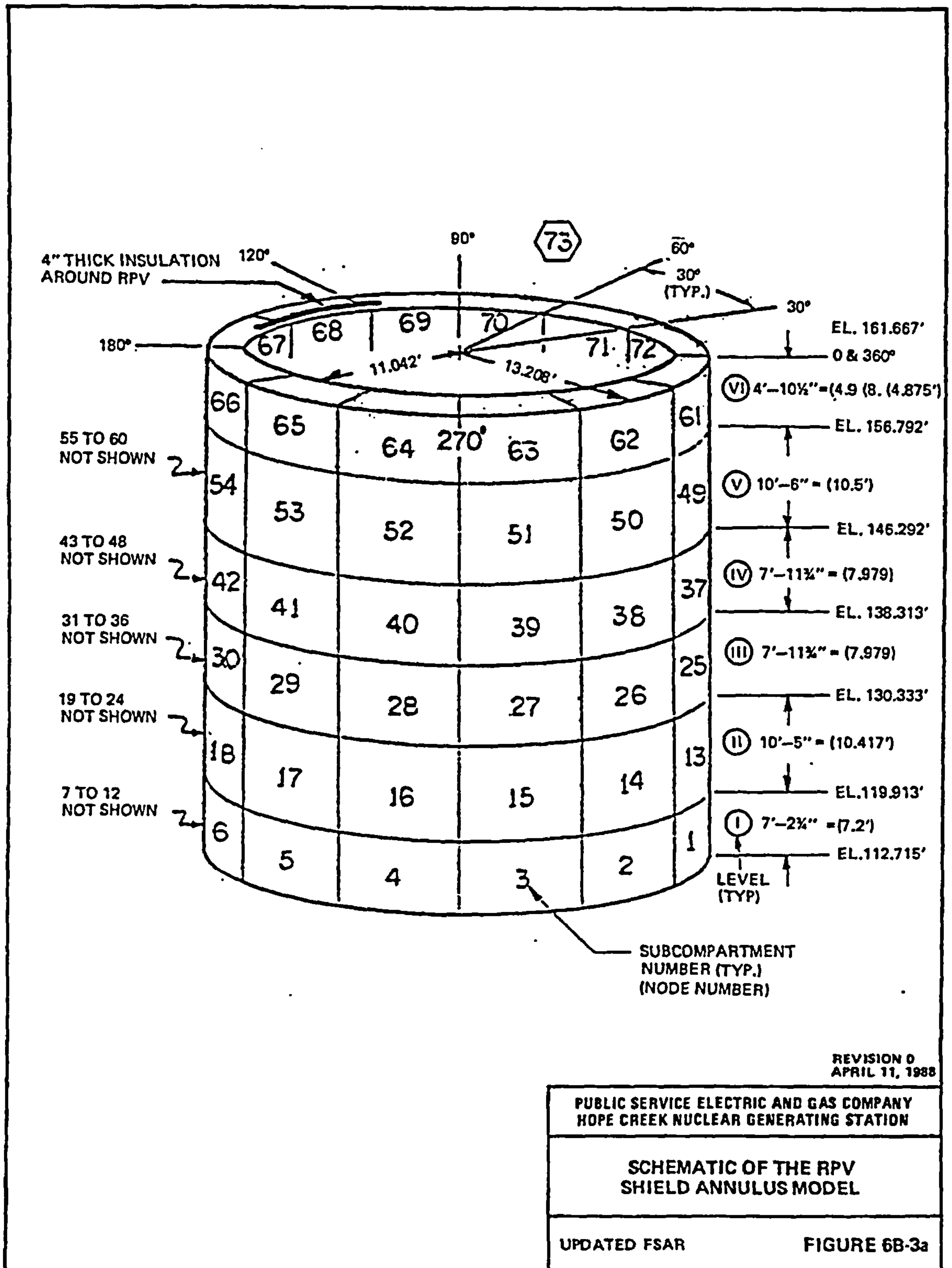

4" THICK INSULATION AROUND RPV
120°
90°
73
60°
30° (TYP.)
30°
180°
67
68
69
70
71
72
11.042'
13.208'
EL. 161.667'
0 & 360°
(VI) 4'–10½"=(4.9 (8. (4.875')
66
65
64
270°
63
62
61
EL. 156.792'
55 TO 60 NOT SHOWN
54
53
52
51
50
49
(V) 10'–6" = (10.5')
EL. 146.292'
43 TO 48 NOT SHOWN
42
41
40
39
38
37
(IV) 7'–11¾" = (7.979)
EL. 138.313'
31 TO 36 NOT SHOWN
30
29
28
27
26
25
(III) 7'–11¾" = (7.979)
EL. 130.333'
19 TO 24 NOT SHOWN
18
17
16
15
14
13
(II) 10'–5" = (10.417')
EL. 119.913'
7 TO 12 NOT SHOWN
6
5
4
3
2
1
(I) 7'–2¾" =(7.2')
EL. 112.715'
LEVEL (TYP)
SUBCOMPARTMENT NUMBER (TYP.) (NODE NUMBER)
REVISION 0
APRIL 11, 1988
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION
SCHEMATIC OF THE RPV
SHIELD ANNULUS MODEL
UPDATED FSAR
FIGURE 6B-3a

Figure 1c-8

d) Provide a design description of annulus region and the existing 25/75 flow diverter which limits the break flow into the annulus to 25% of the nominal value. Where is the postulated break located (i.e., on the recirculation or the feedwater line) and where does the remaining flow go? Is this flow reduction the major contributor to meeting the design requirements? Was the diverter considered in the original code calculations? If not, was a COPDA calculation [done] with the diverter, and what were the results?

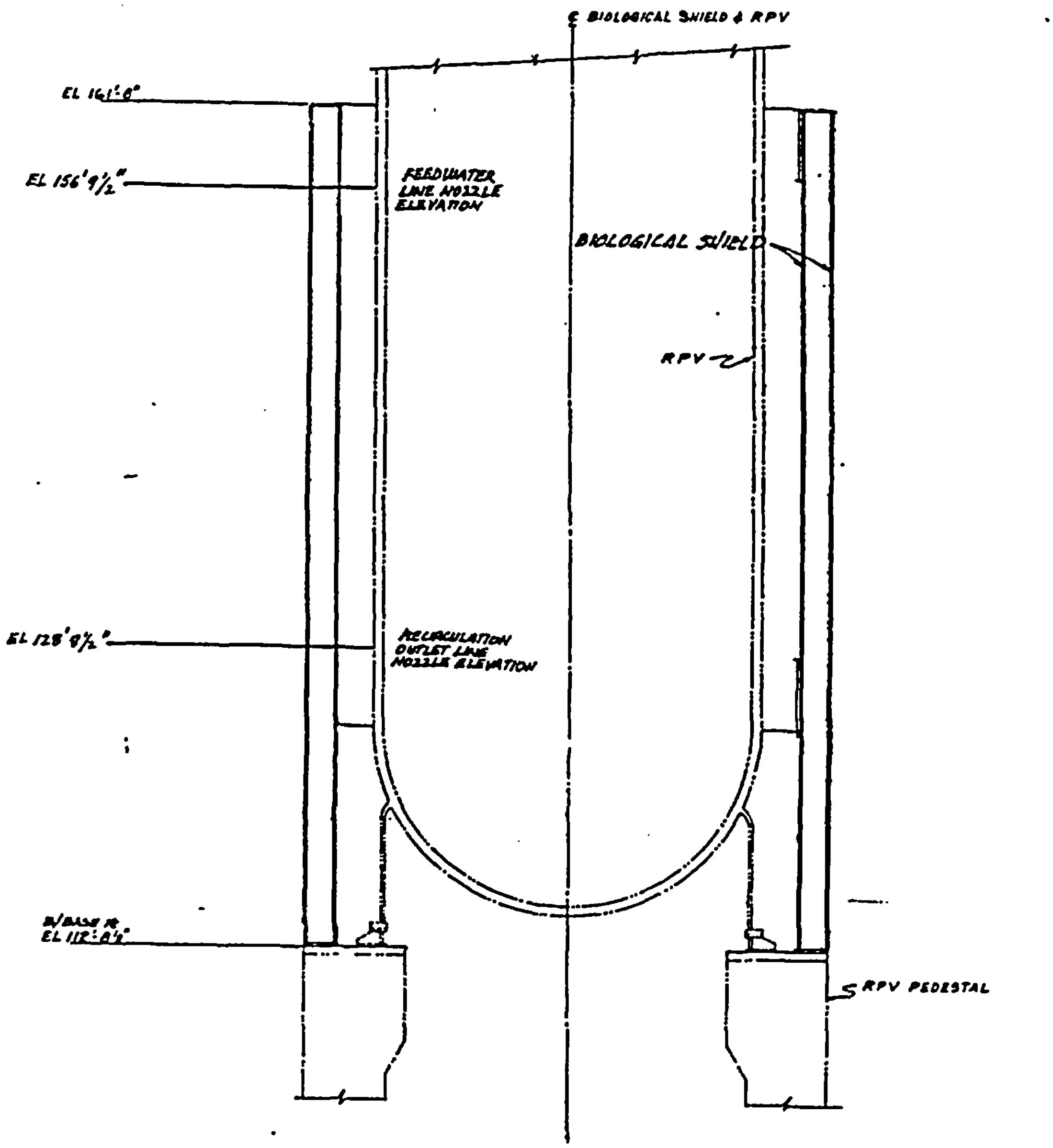
PSEG Response:
The general layout of the annulus region between the reactor pressure vessel and the biological shield wall is shown on Updated Final Safety Analysis (UFSAR) Figure 6B-2, provided here as Figure 1d-1. The flow diverters installed in the shield wall penetrations for the recirculation outlet lines are shown in UFSAR Figure 6B-1, provided here as Figure 1d-2.

PSEG Calculation 12-39 (1979) calculated the thermal hydraulic conditions associated with recirculation and feed water nozzle breaks and loads on the shield wall and reactor vessel. The nozzle RSLB occurs within the shield wall penetration. Due to the physical arrangement and pipe restraints, the double-ended break at the RSLB nozzle would not completely clear the shield penetration opening. Hence, both pipe ends remained within the shield wall penetration opening (i.e., if the break occurs in the shield wall, not all of the mass and energy would enter the annulus area or the drywell area).

Guidance in NEDO-24548 recommended that the flow split be determined based on the ratio of flow areas. Consequently, calculation 12-39 assumed that 50% of the flow would enter the drywell, with the remaining 50% entering the annulus since the flow areas were equal. Using the methodology described in NEDO-24548, calculation 12-39 generated the time history mass and energy releases from a RSLB (the calculation results are reported on Table 6B-1 of the UFSAR). These mass and energy release results were used as inputs to a COPDA calculation to generate pressure-time histories. The time histories were then used to generate response spectra and to establish the dynamic loads on the systems (reactor vessel, reactor internals, shield wall, and the connected piping).

When the shield wall was found to be slightly out of round during construction, calculation 12-79 was prepared to assess the impact of non-symmetric loads on the shield wall. Calculation 12-79 (which also uses the COPDA code) also corrected several discrepancies in 12-39, with the result being that annulus pressures and vessel/support dynamic loads greater than the 12-39 results were predicted. Subsequently, revision 2 to calculation 12-79 (1982) assumed that 25% of the mass and energy flow would be released to the annulus, with 75% to the drywell. The 25/75 flow split was justified by the installation of the flow diverter. This assumption was also in accordance with the recommendations of NEDO-24548 that the flow split be based on a ratio of flow areas. The diverter was calculated to have 859 square inches of opening into the annulus with 2600 square inches of opening to the drywell, for a 25/75 flow split (i.e. 859/(859+2600) = 24.8%).

With the 25/75 flow split, revision 2 to calculation 12-79 (also using COPDA) provided results that were completely bounded by the original design basis loads of calculation 12-39. The net force time history for the COPDA 25/75 flow split is shown on Figure 1c-1. The flow split is a primary contributor to meeting the design basis loads since the 12-79 results would have exceeded 12-39 under the original 50/50 assumption. However, since the flow restrictor did not exist at the time that 12-39 was prepared, the 50/50 assumption is no longer valid and would be an inappropriate assumption under the current configuration of the plant. Therefore, the 25/75 flow split has been part of the Hope Creek licensing basis since revision 2 to calculation 12-79 in 1982, and remains part of the licensing basis under the MELLLA analyses based upon the physical dimensions of the flow diverter.



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HOPE CREEK NUCLEAR GENERATING STATION

REACTOR SHIELD ANNULUS ARRANGEMENT

UPDATED FSAR | FIGURE 6B-2

Figure 1d-1

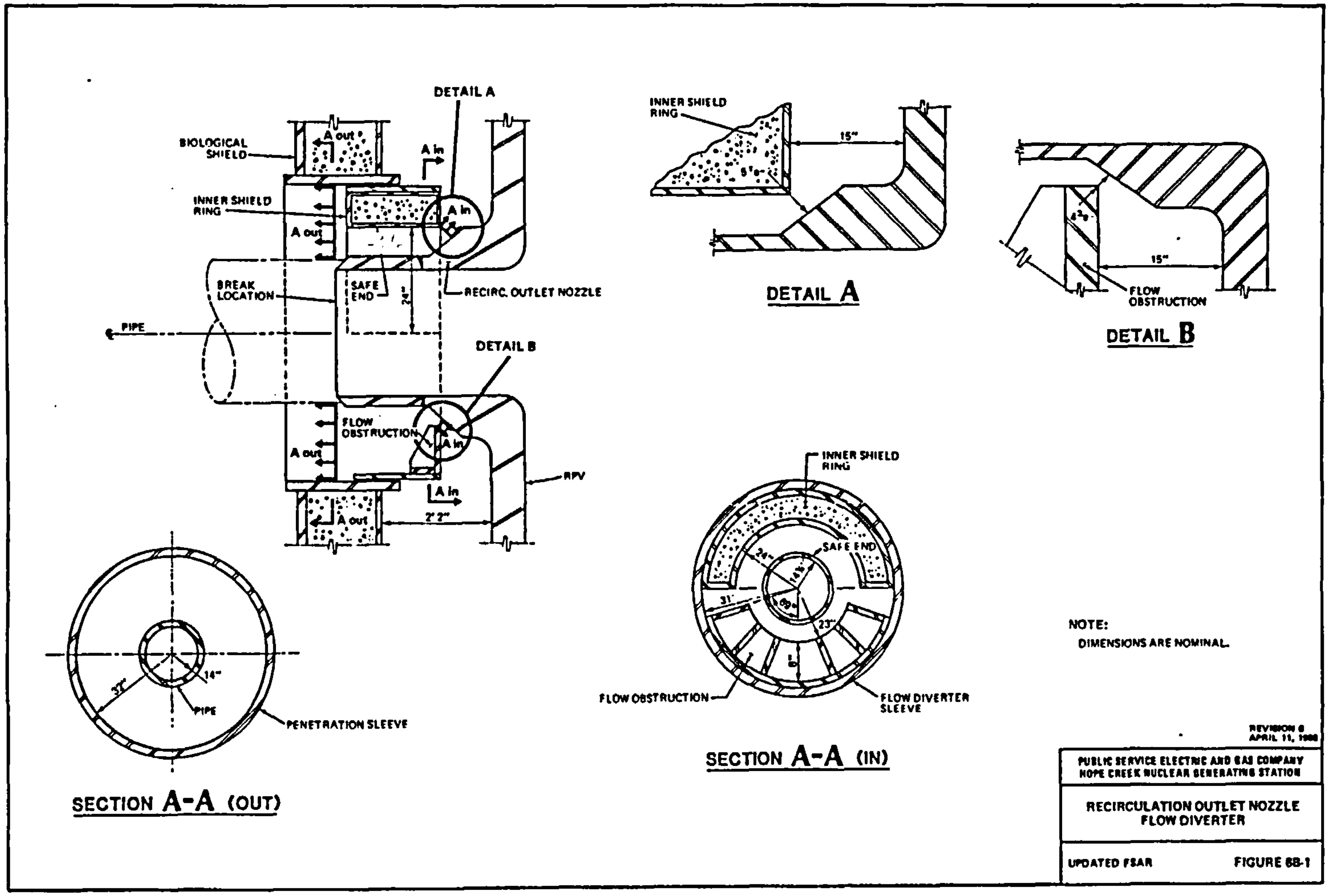
DETAIL A
BIOLOGICAL SHIELD
A out
A in
INNER SHIELD RING
SAFE END
BREAK LOCATION
RECIRC. OUTLET NOZZLE
PIPE
DETAIL B
FLOW OBSTRUCTION
RPV
2' 2"
14"
PIPE
PENETRATION SLEEVE
SECTION A-A (OUT)
INNER SHIELD RING
15"
DETAIL A
15"
FLOW OBSTRUCTION
DETAIL B
INNER SHIELD RING
SAFE END
FLOW OBSTRUCTION
FLOW DIVERTER SLEEVE
SECTION A-A (IN)
NOTE:
DIMENSIONS ARE NOMINAL.
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK NUCLEAR GENERATING STATION
RECIRCULATION OUTLET NOZZLE
FLOW DIVERTER
UPDATED FSAR
FIGURE 6B-1

Figure 1d-2

e) Provide a discussion of the differences in the calculation of the mass and energy releases previously used for the reactor asymmetric loads evaluation (NEDO-24548) to the proposed revised method using the code. Include, for example, the critical flow models used, the equations of state for water used, and the initial primary systems conditions (pressure, temperature, mass, etc.). Provide comparison graphs of the mass and the energy releases, over the time period of interest, for both methods.

PSEG Response:

Critical flow model:	Moody's Slip Flow Model
Equations of state of water:	ASME Steam Table (Fifth Edition, 1967)

Example Case:
Recirculation Line Break at minimum pump speed condition with feedwater temperature reduction

Power:	66.2% of currently licensed thermal power (3339 MWt)
Core Flow:	39.2% of rated core flow (100 Mlb/hr, rated)
Feedwater Temperature:	360.1°F

Initial Conditions:

	NEDO-24548	LAMB Code
Pressure, psia	1034 (1)	1020 (2)
Enthalpy, Btu/lbm	493.6 (3)	498.8 (4)

NOTE:

1. [[
 ]]
2. Dome pressure. [[
 ]] (1028 psia for this case).
3. This is the downcomer enthalpy corresponding to a dome pressure of 976 psia (990 psia at downcomer location). [[
 ]]
4. This downcomer enthalpy is calculated internally by LAMB code.

Blowdown Mass and Energy Release Profile Comparison
Figures 1e-1 and 1e-2 provide comparison plots for the mass and energy releases into the annulus region. Note that the time domain of interest is 0 to 1 seconds for asymmetric loads in the annulus.

[[

]]

Figure 1e-1

[[

]]

Figure 1e-2

f) [Provide] a discussion of the differences between the COPDA and the COMPARE models for the reactor annulus, including nodalization and the treatment of inter-compartment flow paths. Describe the COMPARE sensitivity studies (nodalization, flow paths, time step, etc.) [p]erformed to develop the final model for use in licensing analyses.

PSEG Response:
The COMPARE geometry models developed using the design-basis physical parameters as provided in Appendix 6B of the HCGS Updated FSAR are identical to the COPDA models. The COMPARE model uses the same design basis nodal volumes, initial conditions, junction flow path area, flow loss coefficients, and inertia terms as used by COPDA, reformatted only as needed to be consistent with COMPARE input requirements.

As provided in the design basis results, the arrangement of the pipes in the annulus determines the most representative level division since the pipes constitute the only significant flow restrictions. The time steps selected by sensitivity study for the COMPARE analysis are sufficiently small to assure that changes in calculated results are minor with continued decrease in time step sizes. Time steps used were 1.0E-04 seconds from 0 to 0.01 seconds elapsed time, and 2.0E-04 seconds thereafter until termination of the COMPARE analysis.

g) Provide a discussion addressing quality assurance control, as delineated in Title 10 of the Code of Federal Regulations Part 50, Appendix B, for the COMPARE analyses that are performed to support licensing actions. Include, for example, configuration control, user training and data validation and verification for model development.

PSEG Response:
The reactor annulus subcompartment analysis was performed by Parsons under an approved Nuclear QA Program. This mature, well-defined Quality Management Program provides for an appropriate level of Quality Assurance and verification for nuclear related projects and tasks. The COMPARE analysis is governed by the Nuclear Quality Manual (NQM) that prescribes a Nuclear Quality Management System based on the regulatory criteria and requirements of 10CFR50 Appendix B, ANSI-N45 Series, and ANSI/ASME NQA-1 Standards.

The requirements of the NQM are implemented by procedures in the Nuclear Procedures Manual, which are identified as appropriate to the scope of activities. Configuration control, user training, and data validation and verification for model development are controlled by approved procedures.

The Nuclear Quality Management Program is fully compliant with the requirements of 10 CFR Part 21 and a detailed procedure addresses the documenting, determination, and reportability requirements.

Parsons' QA program is compliant with GE's QA program requirements. The program was audited by GE in 2003 and 2005.

h) On page 10 of your submittal you state that the GENE LAMB code (NEDE-20566P-A) will be used to provide a more realistic blowdown mass and energy release profile. This would be in lieu of the original NEDO-24548 methodology. Is the COMPARE code part of the NEDO-24548 methodology? What methods are used by GE for the AP pressure time history conversion to nodal force time history? What structural model is used?

PSEG Response:
The NEDO-24548 describes a methodology to calculate the mass and energy release; the LAMB code was used in lieu of the NEDO-24548 methodology.

The COMPARE code is not part of the NEDO-24548 methodology. The COMPARE code is used to calculate transient nodal pressures using the mass and energy release as calculated by the LAMB code. The nodal pressures acting on the normal area on the outside of the reactor vessel are then used to calculate the resultant force time histories as described in HCGS Updated FSAR, Appendix 6B. The resulting forces at each node are used by GE as input to their structural analysis to confirm the response of the reactor vessel and vessel internals.

GE used its proprietary computer program GEAPL to convert the postulated recirculation suction line break AP pressure time history to nodal force time history, for use in the structural model.

The Hope Creek plant-unique primary structure model including the reactor building, shield wall, RPV, and the internals, consistent with the original analysis, was used for the structural analysis, using the computer program SAP4G07V.