COMMONWEALTH EDISON COMPANY

TITLE PAGE

CALCULATION NO. NED-M-MSD-49	PAGE 1 OF 8
X SAFETY RELATED	___ NON-SAFETY RELATED

CALCULATION TITLE

Dresden LPCI/Core Spray NPSHA Evaluation
w/o Overpressure - Post DBA-LOCA

EQUIP NUMBER(S)	STATION/UNIT	SYSTEM
2(3) - 1502A/B/C/D 2(3) - 1401A/B	Dresden 2 & 3	LPCI/Core Spray

REV.	CHRON #	PREPARER	DATE	REVIEWER	DATE	APPROVER	DATE
0	198390	Harry Pahlen	2/22/93	Dan Klee	2/22/93	Carl E Dietz	2/22/93

QE-51.D
EXHIBIT B
REV. 3

9303110237 930305
PDR ADOCK 05000237
Q PDR

COMMONWEALTH EDISON COMPANY

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CALCULATION NO: NED-M-MSD-49		REV 0	PAGE 2 OF 8
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COMMONWEALTH EDISON COMPANY

REVISION SUMMARY

CALCULATION NO: NED-M-MSD-49	REV 0	PAGE 3 OF 8
DESCRIPTION OF REVISIONS/REASON FOR CHANGE		

AFFECTED PAGES

PAGES	REV.	DESCRIPTION

QE-51.D
EXHIBIT D
REV. 3

Purpose/Objective:

The Net Positive Suction Head Available (NPSHA) for the Dresden LPCI/Core Spray pumps under post-DBA LOCA conditions was determined in Reference 1. The purpose of this calculation is to examine the effect on NPSHA of removing the drywell/wetwell overpressure assumed in Reference 1 for the cases involving two pump operation, and comparing with NPSH required (NPSHR) to ensure pump protection.

Assumptions/Inputs:

The assumptions and inputs used in Reference 1 are also used for this calculation with the following inputs changed:

1) Torus pressure is atmospheric pressure - 14.7 psia.

2) Maximum torus drawdown at time of analysis is 1 ft. (versus 2.1 ft. in Ref. 1) as provided and justified in Reference 2. This 1.1 ft. reduction in drawdown results in a 1.1 ft. increase in the static head available to the pumps from 13.29 ft. to 14.39 ft.

References:

1) "Dresden LPCI/Core Spray Pumps NPSHA Evaluation - Post DBA-LOCA", Nuclear Engineering Department Calculation No. NED-M-MSD-43 Rev 1.

2) "Dresden LPCI/Containment Cooling System," GE Nuclear Energy letter from S. Mintz to T. L. Chapman dated January 25, 1993.

3) Hydraulic Institute Test Standards, 1988, Centrifugal Pumps - 1.6.

4) Hydraulic Institute Standards, 14th Edition, 1983.

5) General Electric Report No. GENE-770-26-1092 "Dresden Nuclear Power Station Units 2 & 3 LPCI/Containment Cooling System Evaluation," November, 1992

Equations:

Net Positive Suction Head Available (NPSHA) is determined using the following equation:

$$NPSHA = 144 * \frac{(Pt - Pv)}{\rho} + Z - hL \qquad (1)$$

where: Pt = Torus Pressure (psia)

Pv = Vapor Pressure (psia)

Z = Static Head (ft)

hL = suction losses (ft)

ρ = density (lb/ft^3)

Acceptance Criteria:

NPSH required (NPSHR) is determined by the pump vendor through testing as outlined in the Hydraulic Institute Standards (Reference 4, pps. 73-77). Typically, the total head developed by a pump is monitored at a constant flow while the supplied NPSH is reduced. At some point, the developed head begins to drop off as the pump starts to cavitate. Due to the difficulty in determining exactly when the change begins, a value of 3% drop in head is usually accepted as evidence of the onset of cavitation. The NPSH at this point is defined as the NPSH required (NPSHR). Decreasing the NPSH further results in a continuous (but not instantaneous) drop in head as the pump moves into full cavitation (Figure 1).

Due to this inexact method for determining the NPSHR limit, the small performance degradation beyond the 3% drop in head that does not occur instantaneously (Figure 1), and the conservatisms used in Reference 1 and in this calculation, the acceptance limit for this calculation is defined as the following: if the **NPSH available is greater than or equal to the required NPSH (within an error of minus 1% of the NPSHR)**, then pump protection is ensured and the ability of the pumps to perform their safety function is unaffected. It should also be noted that the use of zero torus overpressure is extremely conservative for a post DBA-LOCA analysis as illustrated in Reference 5.

NPSHA Calculations:

Using Equation 1 and the inputs provided above and in Reference 1, the NPSHA is calculated for each of the two pump cases (Table 1). The required NPSH is also provided and the difference between the two is calculated as "Margin".

Summary/Conclusions:

Available NPSH for the LPCI/Core Spray pumps was determined in Reference 1 for post DBA-LOCA torus conditions. In the calculation above, an additional conservative restriction of zero drywell/wetwell overpressure was used to develop a set of NPSHA values for the cases involving two-pump operation. The results in Table 1 indicate that the available NPSH meets the acceptance criteria described above and is therefore adequate to protect the pumps under these conditions.

Case	Total Flow (gpm)	Single Pp Flow (gpm)	Torus Temp (F)	Torus Press (psia)	Static Head (ft)	Specific Volume (ft3/lb)	Vapor Press (psia)	Suction Losses (ft)	NPSHA (ft)	NPSHR (ft)	Margin (ft)	-1% Criteria (ft)
3	10000	5000	168	14.7	14.39	0.01644	5.722	5.87	29.8	30.0	-0.2	-0.3
3A	8916	4458	171	14.7	14.39	0.016457	6.132	4.67	30.0	26.9	3.1	-0.27

Table 1

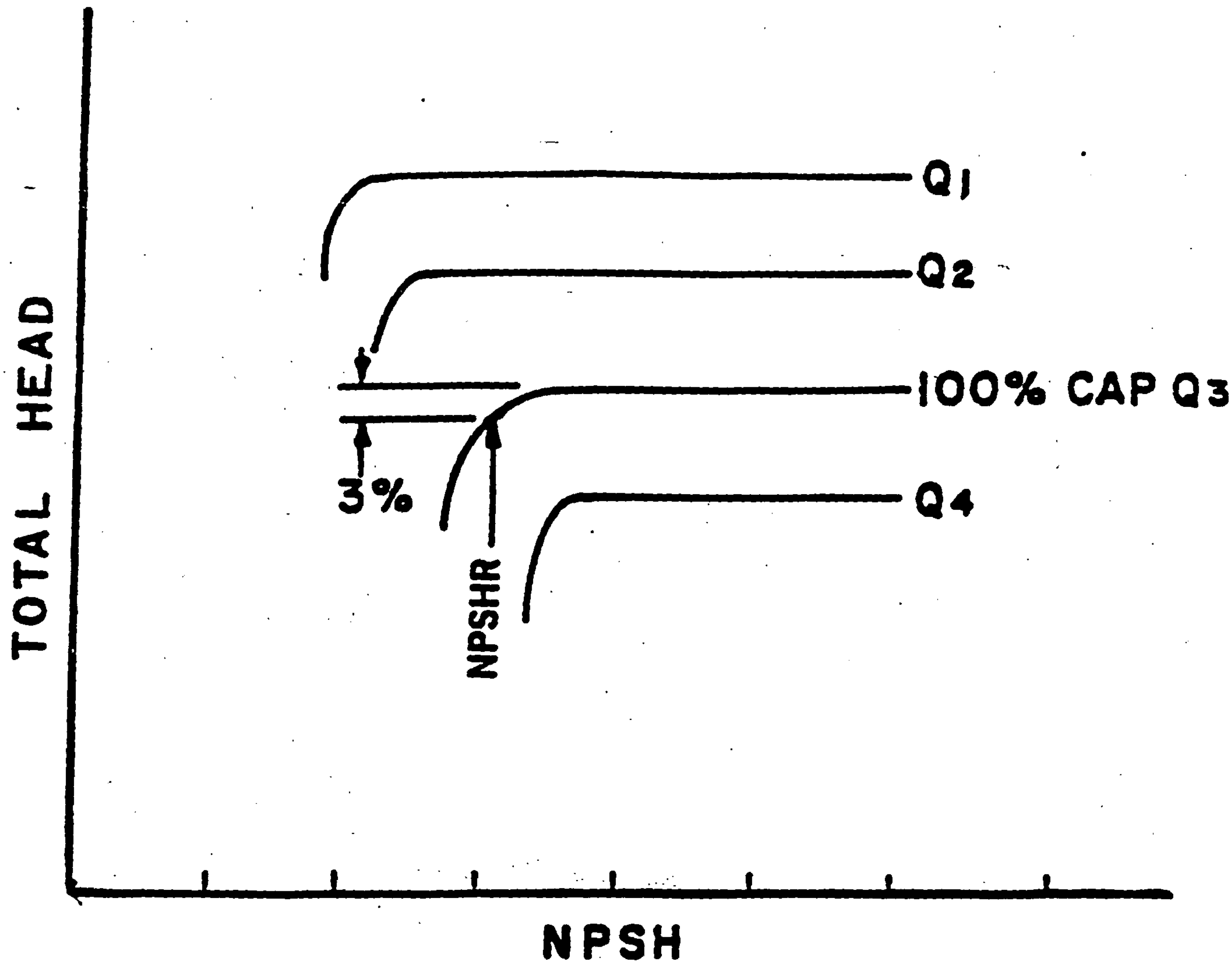


Fig. 1 TYPICAL NPSH TEST

(Reference 3, p. 14)

REVIEW CHECKLIST

CALCULATION NO: NED-M-MSD-49	REV. 0	PAGE 8 OF 8
REVIEWED BY: Dan K Lee		DATE: 2/22/93

YES	NO		REMARKS
☑	☐	1. IS THE OBJECTIVE OF THE ANALYSIS CLEARLY STATED?	
☑	☐	2. ARE ASSUMPTIONS AND ENGINEERING JUDGEMENTS VALID AND DOCUMENTED?	
☐	☑	3. ARE THERE ASSUMPTIONS THAT NEED VERIFICATION?	
☑	☐	4. ARE THE REFERENCES (I.E. DRAWINGS, CODES, STANDARDS) LISTED BY REVISION EDITION, DATE, ETC.?	
☑	☐	5. IS THE DESIGN METHOD CORRECT AND APPROPRIATE FOR THIS ANALYSIS?	
☑	☐	6. IS THE CALCULATION IN COMPLIANCE WITH DESIGN CRITERIA, CODES, STANDARDS, AND REG. GUIDES?	
☑	☐	7. ARE THE UNITS CLEARLY IDENTIFIED, AND EQUATIONS PROPERLY DERIVED AND APPLIED?	
☑	☐	8. ARE THE DESIGN INPUTS AND THEIR SOURCES IDENTIFIED AND IN COMPLIANCE WITH UFSAR & TECH SPECS?	
☑	☐	9. ARE THE RESULTS COMPATIBLE WITH THE INPUTS AND RECOMMENDATIONS MADE?	

10. INDICATE TYPE OF CALCULATION (HAND-PREPARED AND/OR COMPUTER-AIDED) AND METHOD OF REVIEW:

☑ HAND PREPARED DESIGN CALCULATION

THE REVIEW OF THE HAND-PREPARED DESIGN CALCULATION WAS ACCOMPLISHED BY ONE OR A COMBINATION OF THE FOLLOWING (AS CHECKED):

- ☑ A DETAILED REVIEW OF THE ORIGINAL CALCULATION
- ☐ A REVIEW BY AN ALTERNATE, SIMPLIFIED OR APPROXIMATE METHOD OF CALCULATION
- ☐ A REVIEW OF A REPRESENTATIVE SAMPLE OF REPETITIVE CALCULATIONS
- ☑ A REVIEW OF THE CALCULATION AGAINST A SIMILAR CALCULATION PREVIOUSLY PERFORMED

☐ COMPUTER AIDED DESIGN CALCULATION

YES	NO	
☐	☐	11. IS THE PROGRAM APPLICABLE TO THIS PROBLEM?
☐	☐	12. IS THE COMPUTER PROGRAM VALIDATED PER QP 3-64?
☐	☐	13. IS THE COMPUTER PROGRAM VALIDATED BY OTHER AE'S / ORGANIZATIONS AND HAS IT BEEN PREVIOUSLY APPLIED TO NUCLEAR PROJECTS?
☐	☐	14. IS THE INPUT DATA IN CONFORMANCE WITH THE DESIGN INPUTS?
☐	☐	15. ARE THE RESULTS CONSISTENT WITH THE ASSUMPTIONS AND THE INPUT DATA?
☐	☐	16. IS A LIST OF THE PROGRAMS USED AND DATE OF EACH COMPUTER RUN REFERENCED IN THE CALCULATION?
☐	☐	17. IS THE PROGRAM VERSION AND IT'S REVISION IDENTIFIED ON THE COMPUTER RUN?

November 24, 1992

Mr. Brian Viehl

Subject: Recommendations for Tube Replacement versus Plugging on LPCI Heat Exchangers

Reference: CCSW Followup Actions Request for Assistance, Dresden Station, Commonwealth Edison, Brian Viehl, May 1, 1992.

The reference letter requested that Nuclear Fuel Services investigate the relationship between tube replacement with AL-6XN material versus plugging. The intent was to develop an expression and/or graph of the number of tubes that can be replaced with new lower conductivity material while remaining within the design basis 6 percent plugging (of higher conductivity Monel tubes) assumption used to derive the LPCI heat exchanger thermal performance in various pump flow situations.

This work has been completed and is enclosed as an attachment to this transmittal. As a rule of thumb, it has been determined that replacing 13 tubes yields the same reduction in overall heat transfer rate as plugging a single tube. With the design basis of 144 plugged tubes, this allows for a considerable amount of tube replacement without affecting the design. A curve of allowable numbers of replacements versus plugging is provided.

It should be noted that the material thermal properties for the existing and replacement tube materials were taken from information provided by Dresden Station to NFS via fax on 4/3/92. This transmittal consisted of excerpts of an S&L tube replacement modification review document. The thermal conductivities utilized in this analysis were:

Existing tubes (70/30 Cu-Ni)	19.0 BTU/hr-ft-F
Al 6XN tubes (Stainless)	7.9 BTU/hr-ft-F

These values have been reviewed for consistency with other sources and are believed to be conservative for the application presented. In particular, most handbooks present a slightly lower conductivity value for Monel (by approximately 15%), which if used in this evaluation would lead to even higher allowable numbers of tube replacements per tube plugged.

November 24, 1992

This work has been performed and reviewed in accordance with NFS practices for the performance of safety-related work. If you have any questions regarding this matter, please contact K. Ramsden or P. Kong of my staff.

Everett Young
for Terrance A. Rieck
Nuclear Fuel Services Manager

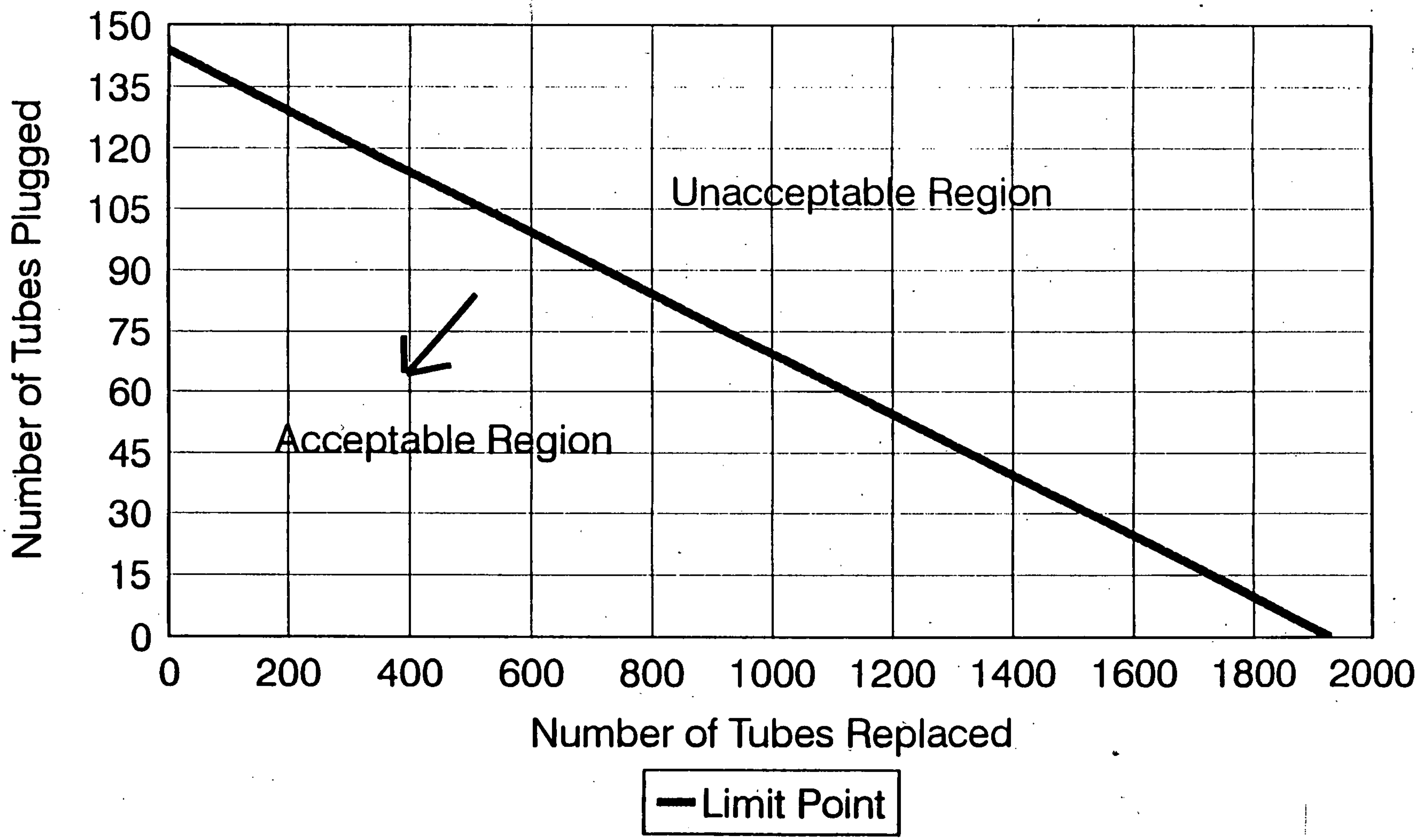
KBR
TAR:KBR:pc

Attachment

cc: NFM-CF
S. Eldridge
G. Lupia
H. Massin
C. Schroeder
B. Wong

Dresden LPCI Heat Exchanger

Tube Replacement with AL 6XN versus Plugging



based on 6% design plugging allowance

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

October 6, 1992

Ms. Sharon Eldridge
Commonwealth Edison Company
Dresden Nuclear Power Station
Rural Route #1
Morris, IL 60450

Subject: LOCA Long-Term Containment Response Analysis K-values for LPCI/Containment Cooling System Heat Exchangers
Dresden Nuclear Power Station, Units 2 & 3

References:
1. Letter, G.G. Chen to S. Mintz, "K Values for Dresden Units 2 & 3 Containment Heat Exchangers," September 14, 1992.
2. Letter, Sharon L. Eldridge/Brian M. Viehl to T. Allen, "Inputs for Heat Exchanger Parameters for CCSW Flow Issue Dresden Units 2 & 3," August 31, 1992.
3. Letter, M.A. Wrightsman to B.M. Viehl, "CCSW Follow-Up Actions, Budgetary Estimate 295-1C3PO-EBO," June 29, 1992.

Dear Ms. Eldridge:

This letter transmits the LPCI/Containment Cooling System heat exchanger K-values calculated by GE (Ref. 1) for the different flow conditions specified in Ref. 2. These K-values will be used to perform the Loss-of-Coolant Accident (LOCA) containment response analysis as described in the "Deliverables" section (for Item 1.) of Ref. 3.

	Case #1 (with loop accuracy)	Case #1 (w/o loop accuracy)	Case #2 (with loop accuracy)	Case #2 (w/o loop accuracy)
LPCI Flow (gpm)	3881	5000	8916	10000
CCSW Flow* (gpm)	3071	3500	4795	5600
K-value (Btu/sec-°F)	219.2	249.6	327.3	356.1

These K-values are based on a service water inlet flow temperature of 95°F and a LPCI inlet flow temperature of 165°F. The K-values are nearly constant as a function of the two inlet flow temperatures for the expected range of inlet flow temperatures. The heat removal rate, Q, of the heat exchanger is calculated using the K-value as follows:

$$Q = K\,(T_{HX,in} - T_{SW,in}) \qquad \text{(Btu/sec)}$$

where $T_{HX,in}$ = Heat exchanger inlet temperature (°F)

$T_{SW,in}$ = Service water inlet temperature (°F).

Please respond with your concurrence of the use of these heat exchanger K-values for the LOCA containment response analysis. If you have any questions on the above, please contact me.

My fax number is (408)925-1674.

*CCSW = Containment Cooling Service Water

Sincerely,

Craig R. Parker

Craig R. Parker
Plant Analysis Services
(408) 925-2025, M/C 469

cc: GE
C.C. Allen
S. Mintz
J.E. Torbeck
DRF T23-00685

NUCLEAR FUEL SERVICES	# of pages ▶ 1
To S. Eldridge	From K. RAMSDEN
Co. CECO	Co. NFS
Dept. -	Phone # 3851
Fax # 131-2922	Fax # 4214

October 13, 1992

Subject: Verification of GE Derived LPCI Heat Exchanger Parameters

To: File

I have run MATHCAD files to develop projected LPCI HX operating points for the flows given to GE by Dresden ENC. GE transmitted their new K-values in the attached fax received on 10/9/92. Using NTU methods with corrected heat exchanger tube lengths from the drawings, values of performance were calculated for the same flow rates as GE. The results are attached, and the values are provided in the following table.

	Case 1a	Case 1b	Case 2a	Case 2b
LPCI flow (gpm)	3881	5000	8916	10000
CCSW flow (gpm)	3071	3500	4795	5600
GE K Value (BTU/sec-F)	219.2	249.6	327.3	356.1
GE HX at 70 degree DT (MBTU/HR)	55.24	62.9	82.48	89.74
NFS NTU K Value (BTU/sec-F)	230.5	263.6	346.8	378.2
NFS NTU HX at 70 degree DT (MBTU/HR)	58.08	66.43	87.4	95.31

The above table values demonstrates that GE is more conservatively calculating the new performance points than would be predicted by the NFS NTU methods. This is consistent with the review of GE's methods performed earlier this year. Therefore we believe that the values developed by GE are appropriately conservative and will result in limiting pool temperature analysis. It appears that GE has about 5-6 percent more conservatism in their methods compared to the NFS NTU approach using public domain heat transfer coefficients.

August 31, 1992

To: T. Allen, General Electric

Subject: Inputs For Heat Exchanger Parameters For CCSW Flow Issue
Dresden Units 2 and 3

As you requested, below are the values for flow and temperatures to be used for your analysis of the Dresden Containments: Please note that two options are given for for each case, one with loop accuracies and one wothout. Calculations for both options are being requested.

- Case 1 (1 LPCI - 1 CCSW)

FLOW	WITH LOOP ACCURACY	WITHOUT LOOP ACCURACY
LPCI	3881	5000
CCSW	3071	3500

- Case 2 (2 LPCI / 2 CCSW)

FLOW	WITH LOOP ACCURACY	WITHOUT LOOP ACCURACY
LPCI	8916	10000
CCSW	4795	5600

These values include instrument loop inaccuracies based on Tech Spec limits and values used for system surveillances. If you have any questions, please call S. Eldridge at Dresden extension 2956.

Sharon L. Eldridge
BWR Site Engineering

Brian Viehl 8/31/92
Brian M. Viehl
BWR Site Engineering
Design Supervisor

cc: K. Ramsden
S. Rhee
NEDCC/Chron Sys Sup

September 9, 1992
In reply refer to
CHRON # 0115517

To: Craig R. Parker, General Electric

Subject: Approval of LOCA Long-Term Containment Response Analysis Input Parameters For Dresden Units 2 and 3.

Reference: 1. GE Letter from Craig R. Parker to S. Eldridge, dated August 7, 1992.

As per the agreement under Shipment Release YY-25 for the above scope of work, Commonwealth Edison has reviewed and approves for use the inputs provided with Reference 1. Please sign the final page and return a copy to me for our records. If there are any additional questions, please call me at (708)-942-2920 extension 2956.

9/9/92
Sharon L. Eldridge
BWR Site Engineering

Brian Viehl 9/9/92
Brian M. Viehl
BWR Site Engineering
Design Supervisor

cc: K. Ramsden - 1/1
NEDCC/Chron Sys Sup

September 21, 1992

Ms. Sharon Eldridge
Commonwealth Edison Company
Dresden Nuclear Power Station
Rural Route #1
Morris, IL 60450

Subject: LOCA Long-Term Containment Response Analysis Input Parameters
Dresden Nuclear Power Station, Units 2 & 3 (Final Values)

Reference: Letter, M.A. Wrightsman to B.M. Viehl, "CCSW Follow-Up Actions, Budgetary Estimate 295-1C3P0-EB0," June 29, 1992.

Dear Ms. Eldridge:

This letter transmits the final signed-off version of the Containment Analysis Input Parameters Verification Form for Dresden Units 2 & 3. This information is to be used to perform the Loss-of-Coolant Accident (LOCA) containment response analysis as described in the "Deliverables" section (for Item 1.) of the referenced letter. The value(s) for "LPCI/Containment Cooling Heat Exchanger K in Containment Cooling Mode" have been determined by GE and will be transmitted to you (with a request for your concurrence) in a separate letter.

If you have any questions on the above, please contact me.
My fax number is (408)925-1674.

Sincerely,

Craig R. Parker

Craig R. Parker
Plant Analysis Services
(408) 925-2025, M/C 469

cc: GE
C.C. Allen
S. Mintz
J.E. Torbeck
DRF T23-00685

August 7, 1992

Ms. Sharon Eldridge
Commonwealth Edison Company
Dresden Nuclear Power Station
Rural Route #1
Morris, IL 60450

Subject: LOCA Long-Term Containment Response Analysis Input Parameters
Dresden Nuclear Power Station, Units 2 & 3

Reference: Letter, M.A. Wrightsman to B.M. Viehl, "CCSW Follow-Up Actions, Budgetary Estimate 295-1C3P0-EB0," June 29, 1992.

Dear Ms. Eldridge:

This letter requests your confirmation of the accuracy of, and agreement with the use of, the containment response analysis input parameters on the attached form. This information is to be used to perform the Loss-of-Coolant Accident (LOCA) containment response analysis as described in the "Deliverables" section (for Item 1.) of the referenced letter. Please fill in the column of values entitled "CECo Proposed", sign the last page under "Verified and Commented by," and return the form to me.

If you have any questions on the above, please contact me.
My fax number is (408)925-1674.

Sincerely,

Craig R. Parker

Craig R. Parker
Plant Analysis Services
(408) 925-2025, M/C 469

cc: GE
C.C. Allen
S. Mintz
J.E. Torbeck
DRF T23-00685

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM FOR DRESDEN UNITS 2 & 3

Parameter	Units	Values: UFSAR	GE Proposed (1)	CECo Proposed	Resolved To Be Used For Containment Analysis
Core Thermal Power	MWt	2527 (2)	2578 (3)		2578
Vessel Dome Pressure	psia		1020 (3)		1020
Drywell Free (Airspace) Volume (including vent system)	ft^3	158236 (4)	158236		158236
Initial Suppression Chamber Free (Airspace) Volume	ft^3				
Low Water Level (LWL)		116300 (4)	120097		120097
High Water Level (HWL)		112800 (4)	116645		116645
Initial Suppression Pool Volume	ft^3				
Max. Water Level		119800 (4)	115655		115655
Min. Water Level		116300 (4)	112203	112000 (25)	112 000

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

Parameter	Units	Values: UFSAR	GE Proposed	CECo Proposed	Resolved To Be Used For Containment Analysis
Initial Drywell Pressure	psig		1.25 (5), (6)		1.25
Initial Drywell Temperature	°F		135 (7)		135
Initial Drywell Relative Humidity	%		20 (8)		20
Initial Suppression Chamber Pressure	psig		0.15 (5)		0.15
Initial Suppression Chamber Airspace Temperature	°F		95 (9)		95
Initial Suppression Chamber Airspace Relative Humidity	%		100 (9)		100
Initial Suppression Pool Temperature	°F		95 (10)		95
No. of Downcomers		96 (4)	96		96
Total Downcomer Flow Area	ft^2		301.6(11)		301.6
Initial Downcomer Submergence	ft				
(HWL)		4.00 (4)	4.00		4.00
(LWL)		3.67 (4)	3.67		3.67

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

Parameter	Units	Values: UFSAR	GE Proposed	CECo Proposed	Resolved To Be Used For Containment Analysis
Downcomer I.D.	ft	2.00 (4)	2.00		2.00
Vent System Flow Path Loss Coefficient (includes exit loss) (12)	-		5.17 (13)		5.17
Supp. Chamber (Torus) Major Radius	ft	54.50 (4)	54.50		54.50
Supp. Chamber (Torus) Minor Radius	ft	15.00 (4)	15.00		15.00
Suppression Pool Surface Area (in contact with supp. chamber airspace)	ft^2			9971.4	9971.4
Supp. Chamber-to-Drywell Vacuum Breaker Opening Diff. Press.					
- start	psid		0.15 (14)		0.15
- full open	psid	0.5 (4)	0.5		0.5
Supp. Chamber-to-Drywell Vacuum Breaker Valve Opening Time (15)	sec		1.0 (16)		1.0
Supp. Chamber-to-Drywell Vacuum Breaker Flow Area (per valve assembly)	ft^2	3.14 (4)	3.14		3.14

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

Parameter	Units	Values: UFSAR	GE Proposed	CECo Proposed	Resolved To Be Used For Containment Analysis
Supp. Chamber-to-Drywell Vacuum Breaker Flow Loss Coefficient (including exit loss)	-		3.47		3.47
No. of Supp. Chamber-to-Drywell Vacuum Breaker Valve Assemblies (2 valves per assembly) (17)		6 (4)	6		6
LPCI/Containment Cooling Heat Exchanger K in Containment Cooling Mode (18)	Btu/s-°F		To be determined by GE.		To be determined later.
LPCI/Containment Cooling Service Water Temperature	°F		95 (19)		95
LPCI/Containment Cooling Pump Heat (per pump)	hp		700 (20)		700
Core Spray Pump Heat (per pump)	hp		800 (21)		800
Time for Operator to turn on LPCI/Containment Cooling System in Containment Cooling Mode (after LOCA signal)	sec	600 (22)	600 (23)		600

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

Parameter	Units	Values: UFSAR	GE Proposed	CECo Proposed	Resolved To Be Used For Containment Analysis
Feedwater Addition (to RPV after start of event; mass and energy)			See attached table.		ok.

GENERAL ELECTRIC CO.
Nuclear Energy Division
ENGINEERING CALCULATION SHEET

NUMBER: DRESDEN - 2/3 DATE: 8-30-78

SUBJECT: F W SYSTEM Fluid/Metal Masses By AMS SHEET ___ OF ___

Component	Fluid Mass (lbm)	Metal Mass (lbm)	Temp. (°F)
RPV			
	34,658	151,935	340
H.P HTR	44,700	162,900	
	51,619	223,854	307
REACTOR F.W. Pump	NEGLIGIBLE	18,000	
	80,851	78,904	305
L.P HTR # 1	64,800	185,100	
	7,000	5,576	229.7
L.P HTR # 2	84,600	246,900	
	15,306	12,191	168.7
L.P HTR # 3	46,800	200,100	
	2,966	2,363	115.9
D.C.	61,800	137,100	
	44,703	52,843	102.2
C.B Pump	NEGLIGIBLE	27,200	
	113,460	89,845	96
Gland St. St. Condenser	7,000	8,130	96
	42,247	28,970	96
SJAE	14,000	28,000	9
	77,120	47,663	96.9
Condensate Pump	NEGLIGIBLE	17,440	96.9
	76,025	35,000	96.9
Hot Well	539,802		96.9

187,457

Verified by
S. Shiga
8-30-78

(see Note 24)

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

<u>Notes</u>:

1. The proposed values are derived from GE-NE documents 22A5743, "Containment Data" (Dresden 2), rev. 1, 4-30-79 and 22A5744, "Containment Data" (Dresden 3), rev. 1, 4-30-79.

2. UFSAR Section 5.2.3.2, p. 5.2.3-4. Note: This corresponds to 100% rated thermal power (see Note 3).

3. GE-NE document, 459HA997, "Heat Balance, Reactor System", rev. 0, 1-21-81 (102% rated thermal power).

4. UFSAR Section 5.2.2, Table 5.2.2:1.

5. GE-NE document, NEDO-24566, "Mark I Containment Program Plant Unique Load Definition, Dresden Nuclear Power Station: Units 2 & 3", rev. 2, April 1982.

6. Based on operating pressure differential of 1.1 psid (see Tech. Spec. Dresden Unit 3, Sect. 3.7, Amend. 75, p. 3/4.7-17).

7. Nominal value.

8. Minimum value.

9. Suppression chamber airspace assumed to be in thermodynamic equilibrium with the suppression pool.

10. Maximum Tech. Spec. value (Dresden Unit 3, Sect. 3.7, Amend. 75, p. 3/4.7-2).

11. $301.6\ ft^2 = \pi \times (2.00\ I.D.)^2/4 \times (96\ downcomers)$.

12. Based on downcomer flow area.

13. GE-NE document, NEDO-21888, "Mark I Containment Program Load Definition Report", rev. 2, Nov. 1981.

14. Not a critical parameter.

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

<u>Notes</u>:

15. Time required for valve assembly to go from a fully closed position to a fully open position with the full-open differential pressure applied across the valve assembly. The full-open differential pressure is defined in the item above. (A valve assembly is two valves in series.)

16. Estimate of maximum opening time.

17. A vacuum breaker valve assembly is two valves in series.

18. $K = Q / (T_{pool} - T_{sw})$

where: Q = heat exchanger heat removal rate (Btu/sec)

T_{pool} = heat exchanger hot-side (supp. pool) inlet temperature (°F)

T_{sw} = heat exchanger cold-side (service water) inlet temperature (°F)

19. UFSAR Section 6.2.4, Table 6.2.4:1.

20. GE Motors document, 992C510, "Outline (Induction Motor)" (LPCI), rev. 4, 1-3-68.

21. GE Motors document, 992C510AB, "Outline (Induction Motor)" (Core Spray), rev. 6, 6-25-68.

22. UFSAR Section 5.2.3.3, p. 5.2.3-12 (One Core Spray and One Containment Spray Cooling Pump Operation case).

23. Standard assumption for operator action time for Mark I containment plants.

24. The feedwater system table is derived from data in the Nutech letter, G.R. Edwards (Nutech) to T.J. Mulford (GE), "Dresden and Quad Cities Containment Data" (Feedwater System - Metal and Liquid Masses, and Temperatures), COM-01-156, August 7, 1978. The actual feedwater addition to the RPV used in the containment analysis is based on the data in the feedwater system table.

25. Minimum Tech. Spec. value (Dresden Unit 3, Sect. 3.7, Amend. 75, p. 3/4.7-1).

CONTAINMENT ANALYSIS INPUT PARAMETERS VERIFICATION FORM
FOR DRESDEN UNITS 2 & 3 (CONTINUED)

Prepared by: Craig R. Parker, Craig R. Parker, Plant Analysis Services — 8-7-92

Signature, Name, Organization, GE Nuclear Energy — Date

Verified and Commented by:

Sharon Eldridge, BWRSD — 9/8/92

Signature, Name, Organization, CECo — Date

Final Values Resolved by:

Craig R. Parker, Craig R. Parker, Plant Analysis Services — 9-21-92

Signature, Name, Organization, GE Nuclear Energy — Date

Sharon Eldridge, BWRSD — 9/8/92

Signature, Name, Organization, CECo — Date

GENERAL ELECTRIC

729E583

PROCESS DIAGRAM

LPCI CONTAINMENT COOLING SYSTEM

3-13-69
EP 1520

P.C.F. 105X904 (DRESDEN II & III) 1520

NOTES:

1. AE IS TO FILL IN ALL EMPTY BLANKS. POINTS LISTED WILL BE BASED UPON AE ARRANGEMENTS.
2. ALL ELEVATIONS ARE WITH RESPECT TO L.P.C.I. PUMP ELEVATION WHICH IS ASSUMED "0".
3. THE MINIMUM PUMP N.P.S.H. AVAILABLE OCCURS DURING MODE C AND MODE F AND MUST BE EQUAL TO OR GREATER THAN 31' AND 37.5' RESPECTIVELY.
4.
5. ELEVATIONS ARE NOT INCLUDED IN ΔP VALUES GIVEN. ELEVATIONS SHALL BE INCLUDED WHEN DETERMINING THE VALUES FOR THE EMPTY DATA BLANKS.
6. THE ΔP LISTED IS THE PRESSURE DROP ACROSS THE NOZZLES ONLY.
7.
8. DASHED LINES INDICATE FLOW DOES NOT PASS THROUGH THESE POINTS.

PRIMARY MODES:

A. ACCIDENT W/RECIRC LINE BREAK ON SIDE II WITH 3 PUMP OPERATION AND ANY ONE STRAINER PLUGGED (20 PSIG).
B. POST ACCIDENT LPCI W/HEAT REJECTION W/RECIRC LINE BREAK ON SIDE II WITH 2 PUMP OPERATION AND 1 STRAINER PLUGGED (0 PSIG).
C. POST ACCIDENT CONT. SPRAY W/HEAT REJECTION WITH 2 PUMP OPERATION AND ANY ONE STRAINER PLUGGED.
D. L.P.C.I. SYSTEM TEST DURING PLANT OPERATION.
E. FUNCTION TEST FOLLOWING SHUTDOWN.
F. ACCIDENT W/RECIRC LINE BREAK ON SIDE II WITH ANY ONE STRAINER PLUGGED AND 2 PUMP OPERATION (0 PSIG).

MODE A — RX PRESS = 20 PSIG

POSITION	1	2 ABC	3 ABC	4 AB	5 A	5 B	6 ABC	7 ABC	8 A	8 B	11 A	11 B	12 B	13 B	14 B	15 B	16 B	17 B	18 B	19 B	30
FLOW, GPM	—	5600	5600	5600	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	←					→	4500	—
PRESS., PSIA	14.7																				34.7
TEMPERATURE °F	—	95	←																→	95	—
MAX. PRESSURE DROP, P.S.I.	ΔP=32					ΔP=45		OVERALL ΔP=[illegible]					ΔP=45		ΔP=45		ΔP=51		ΔP=21.6	OVERALL ΔP=97.9	

MODE B — RX PRESS = 0 PSIG

POSITION	1	2 ABC	3 ABC	4 A	5 A	6 AC	7 AC	8 A	9 A	10 A	11 A	12 B	13 B	14 B	15 B	16 B	17 B	18 B	19 B	30		31 A	32 A
FLOW, GPM	—	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	←									→	10700	—		3000	7000
PRESS., PSIA	17.8																			17.8			
TEMPERATURE °F	—	165	←					→	165	145.4	←							→	145.4	—		95	125
MAX. PRESSURE DROP, P.S.I.						ΔP=[illegible]			ΔP=15	OVERALL ΔP=[illegible]												ΔP=5	

DUTY PER HEAT EXCHANGER 105×10⁶ BTU/HR TOTAL HEAT EXCHANGER DUTY 105×10⁶ BTU/HR

MODE C

[illegible]

DUTY PER HEAT EXCHANGER [illegible] BTU/HR TOTAL HEAT EXCHANGER DUTY [illegible] BTU/HR

MODE D

POSITION	1	2 ABC	3 ABC	5 A	5 B	6 ABC	7 ABC	8 A	8 B	11 A	11 B	24 B	25 B	1
FLOW, GPM	—	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	4500	4500	—
PRESS., PSIA	14.7													14.7
TEMPERATURE °F	—	95	←									→	95	—
MAX. PRESSURE DROP, P.S.I.	OVERALL ΔP=45					ΔP=[illegible]					OVERALL ΔP=45			

MODE E

POSITION	29	29 AC	6 AC	7 AC	8 A	11 A	24 B	14 A	15 A	16 A	17 A	18 A	19 A	30
FLOW, GPM	5666	4633	4633	[illegible]	[illegible]	←						→	3666	—
PRESS., PSIA	14.7													14.7
TEMPERATURE °F	95	←										→	95	—
MAX. PRESSURE DROP, P.S.I.			ΔP=[illegible]			OVERALL ΔP=128.1								

MODE F — RX PRESS. = 0 PSIG

POSITION	1	2 ABC	3 ABC	4 A	5 B	6 AB	7 AB	8 B	11 B	24 B	14 B	15 B	16 B	17 B	18 B	19 B	30
FLOW, GPM	—	5600	5600	5600	[illegible]	5600	5600	11200	←						→	11200	—
PRESS., PSIA	14.7																14.7
TEMPERATURE °F	—	95	←												→	95	—
MAX. PRESSURE DROP, P.S.I.						ΔP=[illegible]			OVERALL ΔP=60.9								

LIMITING LINE LOSSES

MODE	POINTS
A	12 THRU 19
B	NONE
C	1-6 (NPSH)
D	15-25
E	28-29
F	1-6 (NPSH)

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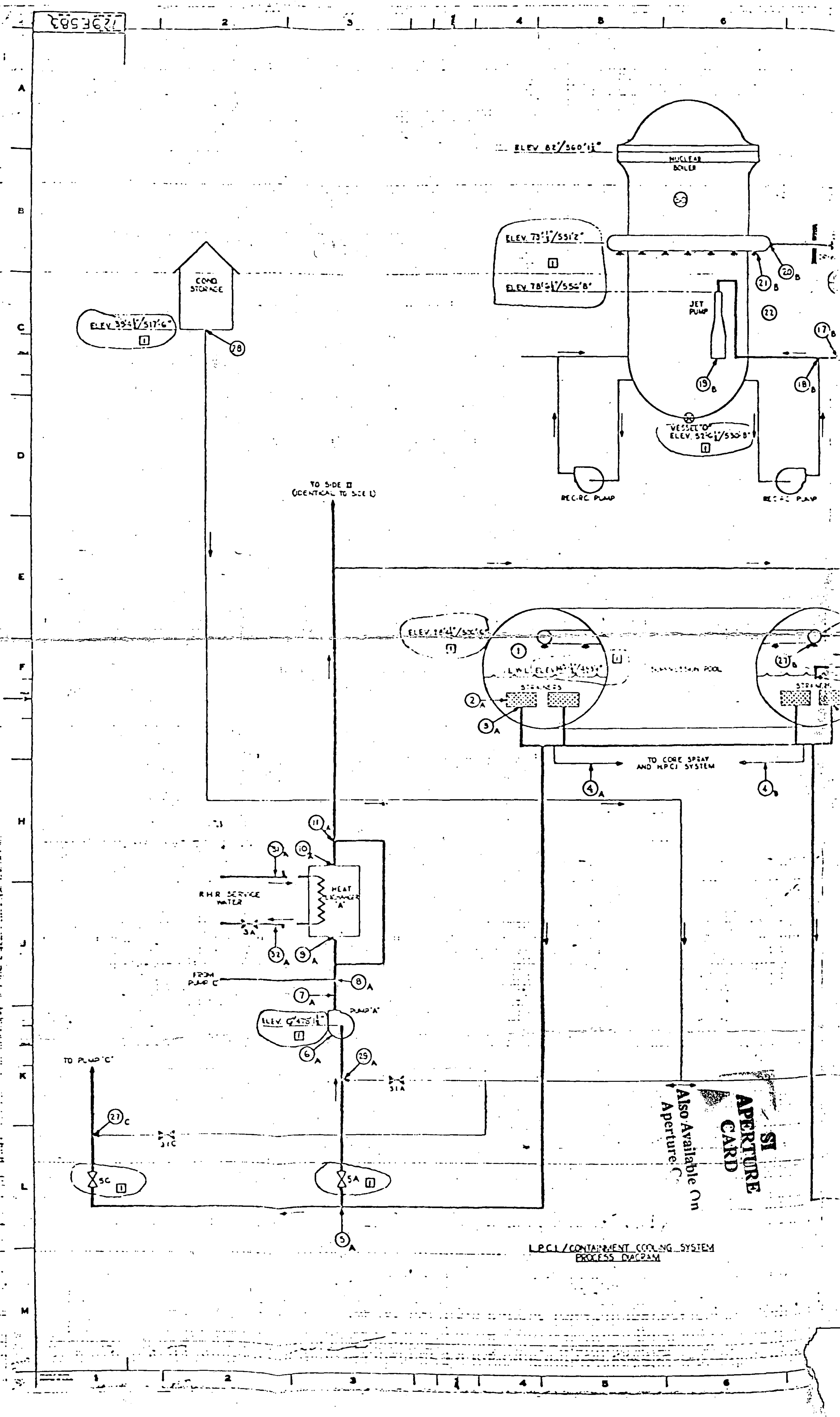

9303110237-02

September 29, 1992
In Reply Refer to
Chron #0115532

TO: C. Schroeder

SUBJECT: Submergence of LPCI Discharge Line Post LOCA
Dresden Units 2 and 3

REFERENCES:

1. Letter from S. Mintz, GE, to S. Eldridge dated September 24 1992
2. Dresden FSAR/UFSAR, Section 5.2.3 for Minimum Submergence of Torus Downcomers of 3.67 ft.
3. Dresden Technical Specification, Section 3.7, Bases, pages B 3/4.7-33 for minimum submergence
4. Drawings M-3302, M-3502
5. CB&I Drawings for Contract No. 9-3600 Numbers: 1, 200, 211
6. Dresden Primary Containment Pathways Evaluation, On Site Review No. 92-42, for Penetrations X-310A and X-310B

At the request of the Station Technical Staff, BWR Engineering has determined the minimum submergence of the LPCI discharge nozzles in the Suppression Chambers, Dresden Units 2 and 3. This evaluation is required to support the information provided in the "Dresden Primary Containment Pathway Evaluation", Reference 6. The conclusion in the above report states that if the nozzle remains covered post LOCA and is therefore not exposed to containment atmosphere, the valves on the associated line are not required to be Local leak Rate Tested (LLRT) per 10CFR50 Appendix J.

From Reference 1 the maximum draw down expected post LOCA is 2.1 feet. Applying this draw down to the low water level as delineated in the Tech Spec and FSAR/UFSAR (References 2 and 3) the post LOCA minimum water level at Elevation 491'-5" provides a minimum submergence of 6" for the subject nozzles. The attached sketch shows the geometry of the respective systems and is a compilation from References 4 and 5.

The information in this evaluation is approved for the Stations use. If you have any other questions, please call me at extension 2956.

Sharon L. Eldridge
BWR Site Engineering

9/29/92

Brian M. Viehl
BWR Site Engineering
Design Supervisor

BMV/SLE/maf (ZENCNE91/98)

cc: M. Strait
M. Andjelic
H. Massin
NEDCC/Chron Sys. Supv.

September 24, 1992

Ms. Sharon Eldridge
Commonwealth Edison Company
Rural Route # 1
Dresden Nuclear Power Station
Morris, IL 60450

Subject: Dresden Units 2 & 3 - Post LOCA Pool Drawdown

References: 1) Telecopy, J.C. Elliott to S. Mintz, "Small Job Authorization," September 21, 1992. (Small Job Task Number DR127)

Dear Ms. Eldridge:

In response to your request and per Reference 1, this letter provides an estimate of the maximum reduction in the Dresden suppression pool level following a loss-of-coolant accident (LOCA). The maximum suppression pool level decrease during a LOCA for Dresden is estimated to be 2.1 feet. This pool level decrease is the same as the value estimated for Quad Cities previously. A comparison of the key parameters between Quad Cities and Dresden confirmed that these plants are essentially identical and therefore allows application of the Quad Cities pool level decrease estimate to Dresden. This pool level decrease can be used together with the lower Technical Specification limit on pool level to determine the minimum post-LOCA pool level.

This estimate of 2.1 feet pool level reduction post-LOCA is based on a conservative analysis of another BWR with a 251 inch ID reactor pressure vessel and a Mark I containment. This analysis considered the water which may be transferred to the reactor pressure vessel, drywell and drywell-to-wetwell vent system from the suppression pool during a LOCA. A more detailed calculation could be performed with the Dresden specific geometry, but it is not expected that the results would change significantly.

Documentation and evidence of verification of this letter is included in the GE design record file, DRF-T23-00692.

Sincerely,

S. Mintz

S. Mintz

cc: <u>GE</u>
J. E. Torbeck
C. T. Young
J. E. Nash

GE NUCLEAR ENERGY
San Jose, CA

PLANT PERFORMANCE ANALYSIS PROJECTS

January 25, 1993

To: T. L. Chapman

From: S. Mintz

Subject: Dresden LPCI/Containment Cooling System

Attachment 1 provides the responses to CECO questions requested by Sharon Eldridge (CECO) in connection to the Dresden LPCI/Containment Cooling System evaluation. Please provide these responses to Sharon Eldridge of CECO.

Documentation and evidence of verification is contained in DRF-T23-685.

S. Mintz

S. Mintz

cc J. E. Torbeck
J. E. Nash
C. T. Young
DRF-T23-685

ATTACHMENT 1

RESPONSE TO CECO QUESTIONS

DRESDEN LPCI/CONTAINMENT COOLING SYSTEM EVALUATION

Question 1:

What is the impact on the peak suppression pool temperature calculated for Case 4a of GENE-770-26-1092 of assuming Containment Cooling Service Water (CCSW) initiation at 1800 seconds for containment cooling instead of the time of 600 seconds used for the GENE-770-26-1092 analysis?

Response to Question 1:

It is estimated that the peak suppression pool temperature for Case 4a of GENE -770-26-1092 will increase by no more than 2°F to no more than 188°F. This temperature increase was estimated by determining the energy removed by the LPCI/Containment Cooling System heat exchanger between 600 and 1800 seconds for Case 4a. This energy was then added to the suppression pool at the time of the peak suppression pool temperature to determine the increase in the peak suppression pool temperature.

It should be noted that this temperature increase is conservative since it is more representative of the increase in the pool temperature at 1800 seconds (i.e., at the new initiation time for CCSW) than the increase in the peak pool temperature. Once CCSW is initiated this increase in the suppression pool temperature will be reduced due to the increased effectiveness of the heat exchanger with a higher suppression pool temperature.

Question 2:

What is the impact of initiation of CCSW at 1800 second (with estimated increase in the suppression pool temperature to 188°F) on the containment pressure at the time of the peak suppression pool temperature relative to the value given in Appendix B of Reference 1 for Case 4a?

Response to Question 2:

At the time of the peak suppression pool temperature the suppression chamber airspace temperature is very close to the water temperature of the containment sprays. A change in the peak suppression pool temperature of 2°F will produce a change in temperature of the spray water temperature of approximately 1°F. Therefore the suppression chamber airspace temperature will also increase by this amount. The change in the suppression chamber vapor pressure for this change in airspace temperature is approximately 0.1 psi. This is therefore the estimated increase in the suppression chamber airspace pressure due to an increase in the suppression pool temperature of 2°F to 188°F.

It should be noted that this estimate conservatively neglects (with respect to minimizing the containment pressure) the effects of the increase in the drywell temperature and drywell vapor pressure with an increase in the suppression pool temperature. An increase in the drywell pressure would result in an additional increase in the suppression chamber airspace pressure due to the flow through the suppression chamber-to-drywell vacuum breakers.

Question 3:

What is the maximum drawdown expected for a LOCA for Dresden?

Response to Question 3:

Reference 2 reported a maximum drawdown of 2.1 feet for Dresden. This was based on a conservative application of pool drawdown analyses conducted for a range of break sizes for another plant with a 251 inch ID reactor pressure vessel and a Mark I containment. However this maximum drawdown (which occurs only for breaks smaller than the DBA-LOCA) would only occur early in the LOCA (< 1000 seconds) when suppression pool temperatures are significantly cooler than the maximum pool temperature. During the times of peak suppression pool temperatures when NPSH is of greatest concern a maximum drawdown of 1 foot is expected.

Question 4:

What is the impact of using May-Witt or ANS 5.1 + 2σ on the peak suppression pool temperature determined for Case 4a of GENE-770-26-1020?

Response to Question 4:

The impact of using May-Witt on the peak suppression pool temperature is estimated based on a comparison of the decay heat near the time of the peak suppression pool temperature. The basis for using this comparison is that at the time of the peak suppression pool temperature the heat rejection rate to the suppression pool from the reactor pressure vessel (RPV) is equal to the rate of containment cooling which is itself established by the suppression pool temperature. The peak suppression pool temperature for Case 4a occurs at approximately 30000 seconds. The value of the May-Witt decay heat is approximately 15% higher than the ANS 5.1 value at this time. This would produce an increase of approximately 14°F in the peak suppression pool temperature. The decay heat for ANS 5.1 + 2σ is less than the May-Witt decay heat therefore the increase with ANS 5.1 + 2σ will be bounded by the 14°F increase estimated for the May-Witt decay heat.

References:

1) GE Report GENE-770-26-1092, "Dresden Nuclear Power Station, Units 2 and 3, LPCI/Containment Cooling System Evaluation," November 1982.

2) Letter, S. Mintz to S. Eldridge, "Dresden Units 2 & 3, - Post LOCA Drawdown," September 24, 1992.

CONTAINMENT ANALYSIS INPUT PARAMETERS

PARAMETERS	UNITS	UFSAR		NEW	JUSTIFICATION
		Section	Value		
Core Thermal Power	MW_t		2,527	2,578	GE-NE document, 459HA997, "Heat Balance, Reactor System", Rev. 0, 1/21/81 - 102% rated thermal power
Vessel Dome Pressure	psia			1020	GE-NE document, 459HA997, "Heat Balance, Reactor System", Rev. 0, 1/21/81 - 102% rated thermal power
Drywell Free (Airspace) Volume (including vent system)	ft^3		158,236	158,236	N/A
Initial Suppression Chamber Free (Airspace) Volume	ft^3				Changed due to conservatism
Low Water Level (LWL)			116,300	120,097	
High Water Level (HWL)			112,800	116,645	
Initial Suppression Pool Volume	ft^3				Changed due to conservatism
Max. Water Level			119,800	115,655	
Min. Water Level			116,300	112,203	

CONTAINMENT ANALYSIS INPUT PARAMETERS

PARAMETERS	UNITS	UFSAR		NEW	JUSTIFICATION
		Section	Value		
Initial Drywell Pressure	psig			1.25	GE-NE document, NEDO-24566, "Mark I Containment Program Plant Unique Load Definition, Dresden Nuclear Power Station: Units 2&3", Rev. 2, April 1982 Based on operating pressure differential of 1.1 psid (Dresden Tech Spec. Unit 3, Sect 3.7, Amend 75, p. 3/4.7-17)
Initial Drywell Temperature	°F			135	Nominal value
Initial Drywell Relative Humidity	%			20	Minimum value
Initial Suppression Chamber Pressure	psig			0.15	GE-NE document, NEDO-24566, "Mark I Containment Program Plant Unique Load Definition, Dresden Nuclear Power Station: Units 2&3", Rev. 2, April 1982
Initial Suppression Chamber Airspace Temperature	°F			95	Suppression chamber airspace in thermodynamic equilibrium with the suppression pool
Initial Suppression Chamber Airspace Relative Humidity	%			100	Suppression chamber airspace in thermodynamic equilibrium with the suppression pool

CONTAINMENT ANALYSIS INPUT PARAMETERS

PARAMETERS	UNITS	UFSAR		NEW	JUSTIFICATION
		Section	Value		
Initial Suppression Pool Temperature	°F			95	Maximum Tech. Spec. value, Section 3.7, Amend. 75, p. 3/4.7-2
Number of Downcomers			96	96	N/A
Total Downcomer Flow Area	ft^2			301.6	This value is equal to π x $(2.00 \text{ I.D.})^2/4$ x (96 downcomers)
Initial Downcomer Submergence	ft				N/A
(HWL)			4.00	4.00	
(LWL)			3.67	3.67	
Downcomer I.D.	ft		2.00	2.00	N/A
Vent System Flow Path Loss Coefficient (includes exit loss)				5.17	GE-NE document, NEDO-21888, "Mark I Containment Program Load Definition Report", Rev. 2, November 1981 S&L Calculation: NSLD-3C2-0978-001, Rev. 0, 12/8/78
Suppression Chamber (Torus) Major Radius	ft		54.50	54.50	N/A
Suppression Chamber (Torus) Minor Radius	ft		15.00	15.00	N/A

CONTAINMENT ANALYSIS INPUT PARAMETERS

PARAMETERS	UNITS	UFSAR		NEW	JUSTIFICATION
		Section	Value		
Suppression Pool Surface Area (in contact with suppression chamber air space)	ft^2			9,971.4	S&L Calculation: NSLD-3C2-0978-001, Rev. 0, 12/8/78
Suppression Chamber-to-Drywell Vacuum Breaker Opening Differential Pressure	psid				
- start				0.15	Not a critical parameter
- full open			0.5	0.5	N/A
Suppression Chamber-to-Drywell Vacuum Breaker Valve Opening Time	sec			1.0	**Estimate**
Suppression Chamber-to-Drywell Vacuum Breaker Flow Area (per valve assembly)	ft^2		3.14	3.14	N/A
Suppression Chamber-to-Drywell Vacuum Breaker Flow Loss Coefficient (including exit loss)				3.47	
Number of Suppression Chamber-to-Drywell Vacuum Breaker Valve Assemblies (2 valves per assembly)			6	6	N/A

CONTAINMENT ANALYSIS INPUT PARAMETERS

PARAMETERS	UNITS	UFSAR		NEW	JUSTIFICATION
		Section	Value		
LPCI/Containment Cooling Heat Exchanger K in Containment Cooling Mode	Btu/s-°F			To be determined by G.E.	
LPCI/Containment Cooling Service Water Temperature	°F			95	UFSAR Section 6.2.4, Table 6.2.4:1
LPCI/Containment Cooling Pump Heat (per pump)	hp			700	GE Motors document, 992C510, "Outline (Induction Motor)" (LPCI), Rev. 4, 1/3/68
Core Spray Pump Heat (per pump)	hp			800	GE Motors document, 992C510AB, "Outline (Induction Motor)" (Core Spray), Rev. 6, 6/25/68
Time for Operator to turn on LPCI/Containment Cooling System in Containment Cooling Mode (after LOCA signal)	sec		600	600	N/A
Feedwater Addition (to RPV after start of event; mass and energy)					See Attachment

S. Khee

May 15, 1992

To: C. W. Schroeder

Subject: CCSW Reduced Flow Design Issues
Dresden Units 2 and 3

References:

1) Letter dated April 7, 1992 from B. M. Viehl to C. W. Schroeder transmitting results of reduced CCSW flow review.

2) Letter # E12-00126 from General Electric to B. M. Viehl, dated May 13, 1992, attached.

3) Letter from T. Rieck, Nuclear Fuel Services Providing acceptance of use of ANS 5.1 Heat Decay Methodology and a Sensitivity Evaluation of the Effect of LPCI Heat Exchanger Performance during Transients, dated May 15, 1992, attached.

4) Letter from S. Powers to B. M. Viehl, dated May 15, 1992, with ENC-QE-81 review of GE letter (reference 2) attached.

The purpose of this letter is to transmit the responses to items 2 and 3 follow-up items from Reference 1, concerning the LPCI Heat exchanger duty Calculations and use of instrument accuracy in these calculations.

Results of Calculation

The original LPCI heat exchanger duty calculation could not be retrieved. A new calculation was performed and resulted in a 9% decrease in heat removal capability for 1 pump/1 pump operation (Mode C). The decay heat methodology used in the original Containment Analysis is not consistent with today's accepted decay heat methodology. Today's decay heat methodology provides approximately a 15% margin. This results in a 15% - 9% = 6% overall positive margin in the Containment Analysis.

Discussion

The analysis (Reference 2) was performed to obtain an appropriate heat exchanger duty for support of the parameters shown on the original LPCI system Process Flow Diagram, GE drawing 729E583, Rev 1. A current analysis was performed using current heat exchanger and decay heat methodologies after a search of both General Electric and Senior Engineering (the current heat exchanger vendor company name) records failed to locate any original design calculations. The analysis shows that the heat

exchangers have sufficient heat removal capability under the reduced flow conditions of 1 LPCI pump and 1 CCSW pump (Mode C) to mitigate the required accident conditions. The calculated heat duty available under 1 pump/1 pump operation is less, but use of the current decay heat methodology from ANS 5.1 provides a greater reduction in the heat input. This combination results in a margin of approximately 6% between heat removal capability and decay heat input. The use of ANS 5.1 for calculation of heat decay under accident conditions has been validated by Nuclear Fuel Services in Reference 3. The calculation results, assumptions and input parameters of Reference 2 have been reviewed by the NED/Mechanical & Structural Design Group (reference 4) in accordance with ENC-QE-81 and are found acceptable.

The General Electric Letter in reference 2, also provides justification for the use of analytical values for the flow rates without specific consideration for instrument inaccuracies. The design philosophy used for the original design basis calculations, provided sufficient conservatisms to obviate the need for specifically accounting for any postulated instrument inaccuracies by maximizing/minimizing the analysis conditions.

In accordance with our proprietary agreements with General Electric, the calculation supporting Reference 2 will be available for review at Dresden after May 18, 1992. The remaining issues from reference 1 concerning the FSAR/UFSAR discrepancies and the maximum number of heat exchanger tubes which can be plugged will be completed by September 1, 1992. If you have any questions concerning this assessment, please call me at extension 2956.

Sharon L. Eldridge 5/13/92
Sharon L. Eldridge
BWR Site Engineering

Brian Viehl 5/15/92
Brian M. Viehl
BWR Site Engineering
Design Supervisor

cc: M. Strait, G. Gates
L. Gerner, H. Massin
B. Rybak, C. A. Moerke
G. Smith, K. Ramsden
W. Groszko, J. Kotowski
NEDCC/Chron System Supervisor

General Electric Company
175 Curtner Avenue San Jose, CA 95125

May 13, 1992
CCA9203
GE-NE-668-07-0592
DRF # E12-00126

Mr. Brian M. Viehl
Dresden Site Engineering Design Supervisor
Nuclear Engineering Department
Commonwealth Edison Company
Rural Route 1
Morris, Illinois 60450

Subject: Design Basis for Low Pressure Coolant Injection / Containment Cooling System Heat Exchanger Sizing

Reference:

1. S. Mintz letter to J. E. Nash dated 4/6/92: Same Subject
2. Brian M. Viehl letter to M. W. Hansen dated 4/14/92: CCSW Follow-up Actions Request for Cost Estimate
3. LPCI Containment Cooling System Process Diagram 729E583, Rev. 1
4. Gordon Chen's letter to Tim Allen dated May 13,1992: Dresden 2 Containment Heat Exchanger Mode C Heat Transfer Capability
5. NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate", June 1991

Dear Brian:

In Reference 1, GE summarized the current understanding of the basis for the Dresden LPCI/Containment Cooling System heat exchanger sizing and associated analyses based on an expedited review of readily available information.

In Reference 2, Commonwealth Edison Company requested a cost estimate on pursuing and closing four open items relating to a CCSW reduced flow concern not fully addressed by Reference 1. Advance approval was obtained to work on question 2 and 3 in order to provide a timely response by May 13, 1992. These questions are repeated below with GE's response.

Question 2:

A copy of the original General Electric heat exchanger duty calculation for modes B and C of the original LPCI system Process Flow Diagram.

Response:

Design Procedures extant at the time the original calculations were accomplished did not require the file retention of calculational backup data to design drawings. Nevertheless, GENE initiated a search of known files and interviewed engineering personnel who may have had knowledge of the original calculational methods. After several days of effort, it became apparent that efforts in this direction would prove fruitless.

In light of the above, GENE elected to use current methodology to establish Containment Cooling HX performance under the conditions shown in "Modes B and C" in GE Drawing 729E583 (Reference 3). This effort involved construction of a heat exchanger performance analysis model from known parameters listed on the original Heat Exchanger Data Sheet and GE approved, fluid heat transfer coefficients. Calculations based upon this new model yielded terminal temperatures and heat transfer rates that were inconsistent with the original Heat Exchanger Data Sheet and with Modes B and C of the above drawing. However, the calculation results showed that the original methods did in fact reduce the value of "U" (Overall Heat Transfer Coefficient) from Mode B conditions to Mode C, in order to reflect the large scale reductions in flow, both on the tube and shell side of the heat exchanger. The results of these new calculations for Mode C flow conditions are shown in Reference 4 attached.

To resolve the above inconsistency between new calculations and the conditions shown on the Heat Exchanger Data Sheet, GENE contacted the original heat exchanger manufacturer (now Senior Engineering). These discussions led to the manufacturer's search of his records, both in Los Angeles, California and in Berlin, Wisconsin. Again, these data recovery efforts were not successful. The manufacturer agreed to run a set of calculations, using its current proprietary codes, in an attempt to duplicate the heat exchange rates and terminal conditions shown on the Heat Exchanger Data Sheet. These efforts also showed inconsistency with the original data sheets, however, the results closely agreed with those produced by GENE methods.

In that the original heat exchanger design methodology was not recoverable and current calculations indicated an approximate 9% difference in heat transfer at the original design conditions, GENE and

CECO elected to confirm the current capability of the original Heat Exchanger. This effort was undertaken to conclusively show that DBA maximum suppression pool temperature margins (for LPCI pump NPSH purposes) are conservative. The effort utilized the results of the more realistic decay heat calculation methodology (ANS 5.1) to compare with the results of the older (May-Witt) method used in the original design of the Dresden units. The use of ANS 5.1 decay heat in evaluations of the long-term containment pressure and temperature response with power uprate was presented to the NRC in a submittal (Reference 5) which has been approved by the NRC. This comparison shows that in the post-DBA time frame of maximum suppression pool temperature, decay heat is at least 15% less than the value used in the original design (84.5E6 BTU/hr). Hence, given the currently calculated heat exchanger heat transfer rate of 77E6 BTU/hr (at Mode C conditions) for the original heat exchanger, and a more realistic decay heat rate generation, the maximum suppression pool temperature, post DBA, would be less than the 180F maximum temperature specified in Mode C of GE Drawing 729E583 (Reference 3). In addition, since Mode C specifies the use of one Containment Cooling Heat Exchanger, one Containment Cooling Service Water pump, and one LPCI pump, it is concluded that this equipment combination is adequate for post-DBA suppression pool containment cooling purposes.

Question 3:

Address how Instrument Inaccuracy was taken into account with the original design analysis.

Response:

Instrument inaccuracy was <u>not</u> taken into account in the original design analysis of the long term pressure/temperature response of the primary containment. The analyses made in that time period did not explicitly consider instrument accuracy although initial conditions used in analyses were normally maximized or minimized to insure conservative results.

Typically when instrument inaccuracy is taken into account and the safety analysis is revised it has been shown that margins to design and safety limits still exist. The evaluation performed in response to question 2 above is typical. Using current methodology there is an estimated 6% margin, even taking into account the lower flow rates of the LPCI/Containment Cooling System Heat Exchangers. Computer analysis for the Emergency Core Cooling System also show similar results. i.e. SAFER and GESTR have much more realistic results and lower values than the original safety analysis performed for Dresden.

Prepared by:

C. C. Allen
Plant Systems Design

Verified by: S. Mintz
Plant Analysis Services

Verified by: 5/14/92
G. G. Chen
Heat Exchanger and Pump Design

Reviewed by:
J. E. Torbeck
Plant Analysis Services

Reviewed by:
R. W. Howard
Plant Systems Design

Attachment: Gordon Chen's letter to Tim Allen dated May 13,1992: Dresden 2 Containment Heat Exchanger Mode C Heat Transfer Capability

cc: S. S. Dua
M. W. Hansen
G. L. Hayes
M. G. McBride
W. G. Myers
R. S. Vij

GE Nuclear Energy

cc: R. L. Hughes

Date: May 13, 1992

To: Tim Allen

From: Gordon Chen
Heat Exchanger & Pump Design

Subject: Dresden 2 Containment Heat Exchanger Mode C Heat Transfer Capability

In response to Dresden site request, the heat transfer capability of Dresden 2 Containment Heat Exchanger under Mode C operation has been calculated. The design data, analyzed conditions and the calculated results are given below:

DESIGN DATA

The following heat exchanger design data obtained from PERFEX Heat Exchanger Specification Sheet were used in performing the calculations.

1) Effective Tube Surface Area = 9,897.3 ft^2 (not including 6% excess tubes)
2) Tube Side Flow Velocity At A Flow Rate Of 3,500,000 lbs/hr = 5.7 ft/sec
3) Shell Side Flow Velocity At A Flow Rate Of 5,350,000 lbs/hr = 5.0 ft/sec
4) Fouling Resistance Inside Tubes = .002
5) Fouling Resistance Outside Tubes = .0005
6) Tubes = 70-30 Cu. Ni. 3/4 BWG 18
7) Tube OD = .75"
8) Tube ID = .652"
9) No. of Tubes Per Pass = 1184 (not including 6% excess tubes)

MODE C CONDITIONS

The following conditions, taken from the Containment Heat Exchanger process diagram, are analyzed for Mode C operation:

	Mode C Conditions
Tube Side Flow	3500 GPM
Shell Side Flow	5000 GPM
Tube Side Inlet Water Temperature	95 °F
Shell Side Inlet Water Temperature	180 °F

CALCULATED RESULTS

Presented below are the calculated heat exchanger performance results for Mode C operation:

	Calculated Values
Overall Coeff. Of Heat Transfer, U, Btu/hr-ft²-°F	189
Heat Transfer Rate, (service), Q, Btu/hr	77.0×10^6
Tube Side Water Inlet Temperature, °F	95
Tube Side Water Outlet Temperature, °F	139.29
Shell Side Water Inlet Temperature, °F	180
Shell Side Water Outlet Temperatura, °F	148.35

The calculations for the above heat exchanger performance are contained in DRF No. A00-05269.

Prepared By: G. G. Chen 5/13/92
G. G. Chen, Principal Engineer
Heat Exchanger & Pump Design

Verified By: B. F. Price 5/13/92
B. F. Price
Heat Exchanger & Pump Design

May 15, 1992

Mr. Charles W. Schroeder

Subject: Sensitivity Calculation on LPCI Heat Exchanger Performance during Transients

References: 1. Letter, T. Rieck to C. Schroeder dated 4/7/92 transmitting RSA-D-92-01 , "Evaluation of Reduced CCSW Flow at Dresden Station".

2. Telephone conversation, NFS and GE on 5/12/92.

3. Microfiche of RETRAN Calculations, NFSKRB J(2575) and NFSKRC J(2632).

NFS has been requested to provide an assessment of the impact of the revised LPCI heat exchanger statepoint recently calculated by General Electric. The statepoint provided by GE (Reference 2) was:

LPCI flow (shell side)	5000 gpm
CCSW flow (tube side)	3500 gpm
Torus Temperature	180 F
CCSW Inlet Temperature	95 F
Heat Exchanged	76.9 MBTU/HR
Heat Exchanged per Degree F	0.9047 MBTU/HR-F

The statepoint calculated and used by NFS in Reference 1 for evaluation of the pool heatup transient was:

LPCI flow (shell side)	4500 gpm
CCSW flow (tube side)	3450 gpm
Torus Temperature	165 F
CCSW Inlet Temperature	95 F
Heat Exchanged	63.63 MBTU/HR
Heat Exchanged per Degree F	0.909 MBTU/HR-F

The relatively close heat exchanger performance values are principally a result of very conservative selection of heat exchanger flow rates used in the original NFS calculation. The limiting transient from Reference 1 was re-performed utilizing the heat exchanger performance values provided. The detailed output is contained in Reference 3, appended to the NFS file version of Reference 1. As anticipated the impact was negligible, as demonstrated in the following table.

May 15, 1992

	Heat Exchange Rate	Peak Temperature
Original Calculation	0.909 MBTU/HR-F	167.019
Revised Calculation	0.9047 MBTU/HR-F	167.055

Since the revised calculation results in less than 0.1 F change in the predicted bulk pool temperature, the conclusions of the original calculation remain valid. The plant can effectively mitigate the limiting pool heatup transient with a single loop of pool cooling with 1 CCSW and 1 LPCI pump. The local pool temperature limits remain satisfied, and condensate stability of the T-Quenchers is assured.

The site engineering personnel had additional questions concerning the use of the ANS 5.1 1979 decay heat curves. General Electric indicated that the May-Witt curves were originally utilized for the long term post-LOCA suppression pool heatup calculations. These curves are known to contain significant margin relative to the 1979 ANS curves. This is a result of the original uncertainty in decay heat prediction (+/- 20%) applied in the 1973 curves, versus the reduced uncertainty for the 1979 curves(+/- 2%). The attached figure graphically illustrates the reduction in uncertainty associated with the later standard. The later decay heat standards have been applied to a variety of recent analyses and are generally acceptable to the NRC. The exception is in the area of 10CFR 50.46 LOCA analysis, where a 120% decay heat multiplier is proscribed, except for best estimate LOCA applications where nominal values are utilized with uncertainties addressed in statistical combination methods.

If you have any questions regarding this matter, please contact K. B. Ramsden (x-3851) of my staff.

Bruce Palagi for
Terrence A. Rieck
Nuclear Fuel Services Manager

KBR
TAR:KBR:pc

Attachment

cc: NFM-CF
G. P. Wagner
B. M. Viehl
S. Eldridge
K. Kovar
S. Powers

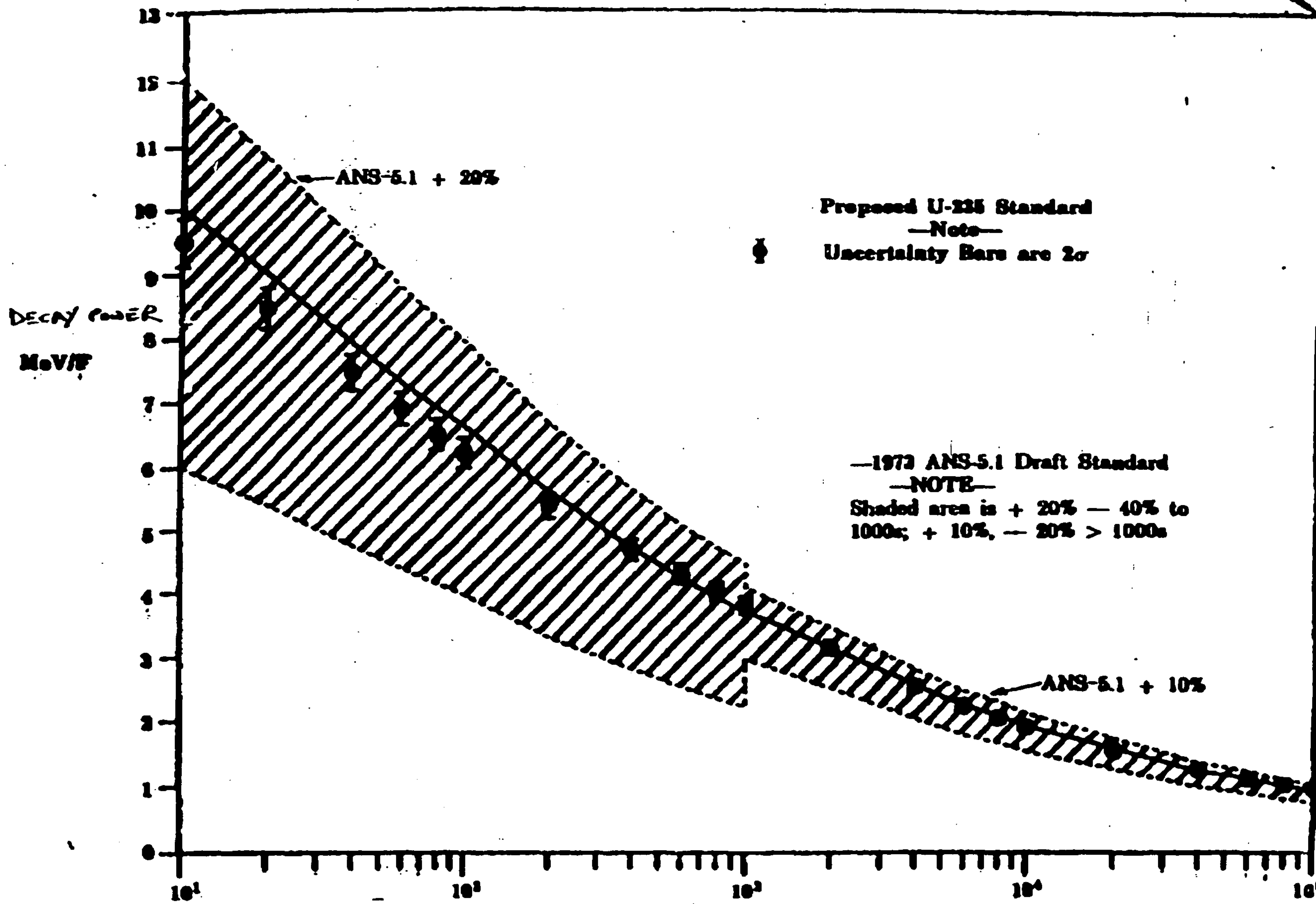

Figure F-1. Comparison of Revised Standard $F(t,\infty)$ for ^{235}U (1979) with 1973 Standard

May 15, 1992
In reply refer to
CHRON # **185905**

To. B. Viehl
ENC- Site Engineering

Subject: **Dresden Unit 2 LPCI Heat Exchanger Mode C Heat Transfer Capability**

Reference: Letter dated 5/13/92 prepared by C.C. Allen (General Electric) to B.M. Viehl

The Nuclear Engineering Department (NED-M&S Group) was requested to evaluate the Heat Exchanger duty calculation for the LPCI Heat Exchanger Mode C (1 CCSW and L LPCI Pump) . General Electric did not have a calculation available for review. However, their results were summarized in Reference 4 of the above reference letter. In accordance with Nuclear Engineering Department procedure ENC-QE-81, a preliminary calculation was prepared to adequately assess the LPCI heat exchanger capacity results supplied by General Electric.

The calculation method employed used the LMTD (Log Mean Temperature Difference) approach because inlet shell, tube temperatures and flows ,and U (heat transfer coefficient) was specified. The preliminary results yielded a value of 78.7 MBTU/HR heat capacity or 2.15% higher as compared with the GE calculated value of 77.0 MBTU/HR. NED feels that this slight difference is negligible and that based on the design inputs their calculated value of 77.0 MBTU/HR is acceptable.

This confirmatory calculation will be formally transmitted in accordance with Nuclear Engineering Department procedure ENC-QE-51.d.

If there are any further questions please contact me on extension 7666, Nuclear Engineering Department.

Prepared by: Samuel P. Powers Jr. Date: 5/15/92
S.P. Powers
Mechanical & Structural Design
Engineer

Approved by: Paul R. Donavin Date: 5/15/92
P.R. Donavin
Mechanical & Structural Design
Supervisor

cc: H.L. Massin (BWR Systems Design)
S. Eldridge (Dresden Site Engineering)

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GE NUCLEAR ENERGY
San Jose, CA

February 17, 1993

Ms. Sharon Eldridge
Commonwealth Edison Company
Dresden Nuclear Power Station
Rural Route No. 1
Morris, IL 60450

Subject: Recommended Use of ANS 5.1 in Dresden 2/3 Containment Long-Term Post Accident Analyses.

References:

1) NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," June 1991.

2) Letter, W. T. Russell (NRC) to P. W. Marriott," Staff Position Concerning General Electric Boiling Water Reactor Power Uprate Program (TAC No. 79384)," September 30, 1991.

Dear Ms. Eldridge

Attachment 1 is a technical description of the use of the GE Decay Heat Model in the subject containment analysis (GE report GENE-770-26-1092). This model is founded on the industry accepted decay heat curve of ANSI/ANS 5.1- 1979 (ANS 5.1). We believe that the attached technical description of the decay heat model and conservatisms in the subject containment analysis relative to the original analysis justifies its application. These conservatisms are confirmed by a comparison of the results of Case 4 of GENE-770-26-1092 to the original FSAR results. Case 4 assumed the same pump flow rates used in the original FSAR analysis for the 1 LPCI/Containment Cooling Spray pump /1 CCSW pump configuration . The results of Case 4 matched the original analysis value of the peak suppression pool temperature of 180°F. This agreement was observed even though the net effect of an approximate 15% reduction in decay heat and a 9% reduction in the Dresden LPCI/Containment Cooling System heat exchanger performance will by themselves result in a lower peak suppression pool temperature. Known conservatisms in the current analysis relative to the FSAR analysis include a 5°F higher initial suppression pool temperature and addition of all feedwater which can contribute to heating the suppression pool.

Information forwarded to GE from Commonwealth Edison Company (CECO), (NRC Letter EA No. 93-019) indicates that CECO is being notified of an apparent

violation due to the use of the ANS 5.1 decay heat model in the subject analysis. The alleged violation (E.3) appears to be based upon the NRC not having approved the decay heat model nor the computer methodology in the analyses. GE Nuclear Energy (GENE) has used the same or similar methodology on several occasions in reports generated for other utilities. On at least one of these occasions the report was reviewed and approved by the NRC.

GENE has had a number of technical discussions with the NRC (NRR) Staff concerning GENE's use of ANS 5.1 and other computer methodology used in analyses similar to that performed for Dresden. GENE and two domestic utilities met with the NRC technical staff in Rockville Maryland in December of 1991 to discuss the GENE containment analysis approach, including computer codes, the use of ANS/ANSI 5.1-1979 (ANS 5.1) decay heat and other assumptions. Although the GE generic power uprate program review was the motivation for the meeting non-power uprate applications were also discussed. The NRC agreed that the GE codes and input assumptions would be generally acceptable for power uprate and other applications, providing that sufficient justification was provided to assure the overall conservatism of the conclusions. It was our understanding that a detailed explanation of the current analysis was to be available for review by the NRC if requested.

The power uprate generic licensing topical report (Reference 1) specifically identified the ANS 5.1 decay heat assumption. That report was approved by the NRC (Reference 2). Several subsequent meetings and phone calls between GE and the NRC have provided GE assurance that the use of GENE containment analysis methods, including computer codes and input assumptions such as ANS 5.1 decay heat, would be acceptable to the NRC technical staff for power uprate or non-power uprate analyses provided the overall conservatism of results was demonstrated.

At no time has the NRC indicated to GENE that prior NRC approval for specific applications was required. Hence, in using this methodology for the Dresden 2/3 analyses we were acting in the belief that we were within known NRC mandates. To date GENE is not aware of any notification by the NRC that the use of ANS 5.1 or GENE computer methods used in the subject analyses are invalid.

Evidence of verification for Attachment 1 is contained in the GENE design record file; DRF T23-00685.

S. Mintz

S. Mintz
Plant Performance Analysis Projects
(408)-925-1791 M/C 469

cc J. E. Torbeck (GE)
J. E. Nash (GE)
T. L. Chapman (GE)
C. T. Young (GE)
R. T. Hill (GE)
D. J. Robare (GE)
DRF-T23-00685

Attachment 1 - Use of ANS 5.1 Decay Heat Model in Containment Analysis

The General Electric analysis of the Dresden long-term containment pressure response for the limiting LOCA event which is described in Reference 1 used a decay heat model based on the ANS/ANSI 5.1 - 1979 (Reference 2) decay heat model. This decay heat model was chosen since it is the current model approved by the American National Standards Institute and both the industry and the NRC have acknowledged that ANS 5.1 standard is a more realistic model than previous models. Although the ANS 5.1 decay heat model is more realistic than that used for the original Dresden FSAR analysis, conservative results for the Dresden containment heatup analysis were assured by use of conservatism in the input and analysis assumptions. The following discussion provides more detail on the use of the ANS 5.1 decay heat model.

1. Background and Description of ANS 5.1 Decay Heat Model

The American National Standards Institute approved and the ANS published the new revised ANS 5.1 standard "American National Standard for Decay Heat Power in Light Water Reactors", (Reference 2). This standard includes significant technical advantages over the older standards in that it deals in great detail with the physics involved and has a significant data base. To use these technical advantages, the General Electric Company developed a generic decay power curve based on the ANS 5.1 standard to provide a more accurate assessment of the decay heat during a LOCA. General Electric used the decay heat correlation developed from the ANS 5.1 decay heat model in ECCS analysis to determine expected results based on realistic methods (Reference 3). These realistic methods were used to establish real safety margins. This methodology is described in Reference 3 which has been reviewed and approved by the NRC.

The generic decay power curve developed by GE, which is based on the ANS 5.1 decay heat model, is described in detail in Appendix B of Reference 3. This decay power curve not only includes fission product decay heat but also

includes other major contributors to post-LOCA heat generation. The other contributors include decay of actinides, decay of activation products, and fission heat due to delayed neutrons. Additional details from Appendix B of Reference 3 are provided below on the major contributors to the decay heat.

Decay Heat from Fission Products

The fission product decay heat is based on the ANS 5.1 decay heat model. In the ANS 5.1 standard, values are provided for decay heat power from fission products from fissioning of the major fissionable nuclides present in Light Water Reactors (LWRs), i.e., U^{235}, Pu^{239} thermal and U^{238} fast. A method is also prescribed for evaluating the total fission product decay heat power from the data given for these specific nuclides. There are fissions produced from other nuclides; however, it is assumed, as directed in ANS 5.1, that all nuclides other than Pu^{239} and U^{238} have the same fission product characteristics as U^{235}.

Decay of Actinides

Actinides are the heavy elements produced from neutron capture in uranium and plutonium isotopes. The actinide concentration following shutdown is calculated assuming that at shutdown the actinide concentration of each actinide is at its equilibrium value. This equilibrium value is determined assuming no captures in the radioactive nuclide. These assumptions conservatively result in a higher actinide concentration.

Decay of Activated Structures

The principal structural material in the reactor is zirconium and is therefore the principal source of decay heat from activated structures. Other materials in and around the reactor are the steel in the control blade, the shroud, and the bottom support plate. However, the contribution to the total decay heat from the activation of control blades or materials outside the core is negligible and is therefore not included.

- Fission Heat Induced by Delayed Neutrons

  When a reactor shuts down, the power level does not drop to zero immediately. Instead it decays away with time due to fissions caused by delayed neutrons. The contribution from delayed neutrons is conservatively determined for LOCAs since it was calculated assuming a slow blowdown rate which results in a smaller void negative reactivity feedback and hence a slower decrease in the neutron flux following the break.

Based on the description given above it was determined that the ANS 5.1 decay heat model provides a more accurate representation of the decay heat during a LOCA than previous models. Additionally, it is noted that the ANS 5.1 standard was conservatively applied in developing the GE decay heat power curve. Therefore there are conservatism retained in the resulting decay heat model.

2. Application of ANS 5.1

Although a more realistic decay heat was used in the Reference 1 analysis, the results are still conservative. The limiting values of key input parameters were used for the Reference 1 analysis. Considering the unlikelihood that each parameter will be at its limiting value at the time of the LOCA this results in very conservative results. It was therefore decided to use a more realistic decay heat model than that used in the original FSAR analysis since the remaining conservatism in the analysis and input assumptions ensure that the calculated pool temperature response is conservative. The conservatism in the Reference 1 analysis are reflected in the following input assumptions:

1) The reactor is assumed operating at 102% of rated thermal power.

2) Feedwater flow into the vessel continues until all the feedwater which will increase the suppression pool temperature is injected into the vessel. In addition, a conservative calculation of the energy in the feedwater piping is added to the RPV/containment system.

3) The initial suppression pool volume is at the minimum Technical Specification (T/S) limit to maximize the calculated suppression pool temperature.

4) The initial suppression pool temperature is at the maximum T/S value to maximize the calculated suppression pool temperature.

5) Service water temperature is assumed to be at the maximum expected value.

6) Passive heat sinks in the containment and in the suppression pool are neglected.

7) All ECCS and RHR system pumps have 100% of their horsepower rating converted to a pump heat input which is added to the RPV liquid or suppression pool water.

8) Heat transfer from the primary containment to the reactor building is conservatively neglected.

10) Heat exchanger performance was calculated based on design fouling factors and assumed 6% plugging of tubes.

11) As requested by CECo, uncertainties in the flow rate to the LPCI/Containment Cooling System Heat Exchanger were considered and lower bounds on the flow rates were used in determining heat exchanger performance.

3. Other General Electric Applications of ANS 5.1 Decay Heat for Containment Analysis

General Electric has used the ANS 5.1 decay heat model in several evaluations and analyses of the containment pressure and temperature response for other BWR plants. Although a more realistic decay heat was used in these analyses, in all cases conservatism were retained in other key input parameters to ensure conservative results.

4. Summary

General Electric used the ANS 5.1 decay heat model in the Reference 1 analysis and in other similar analyses because it is the current standard and is considered a superior model. Although the ANS 5.1 decay heat model is more realistic than the model used for the original Dresden FSAR analysis the calculated suppression pool temperatures are still conservative due to the conservatism retained in the inputs and analysis methods.

5. References

1) GENE-770-26-1092, "Dresden Nuclear Power Station, Units 2 and 3, LPCI/Containment Cooling System Evaluation," November 1992.

2) "Decay Heat Power in Light Water Reactors," ANSI/ANS 5.1 - 1979, Approved by American National Standards Institute, August 29, 1979.

3) General Electric Co., "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," NEDO-23785-1-A, Volume III, October 1984.

CONTROL OF EFFLUENT RELEASES

As discussed during the Enforcement conference on February 22, 1993, protection against effluent releases in excess of 10 CFR 100 limits is ensured by 1) providing a leak tight heat exchanger and 2) providing a positive differential pressure across the LPCI heat exchanger (CCSW outlet pressure > LPCI outlet pressure).

The FSAR/UFSAR includes a description of the design criteria requirement of a positive differential pressure across the LPCI heat exchanger. This dP is ensured by maintaining the operating margin of 20 psid by throttling of the CCSW outlet valve or by increasing LPCI flow (as described in Section 6.2.4.2.4 #3). The original 20 psid value is based on a nominal 5 psid to ensure a positive differential with an additional 15 psid to account for instrument and control inaccuracies(see Attachment A). The "Dresden Station Unit 2 and 3 Setpoint Error Analysis", dated December 28, 1992 has determined that the dP instruments associated with monitoring this dP have a maximum error of 3.26 psid. Therefore to achieve the design criteria requirement a dP of 8.26 is adequate.

The Technical Specification Amendment transmittal dated November 4, 1974 provided an analytical basis for reducing the CCSW discharge pressure requirement to 180 psig. This calculation assumed a LPCI discharge pressure of 125 psig, a 20 psid across the heat exchanger, 10 psig for margin, and a 25 psig pressure drop in the CCSW system from pump discharge to outlet of the heat exchanger. The 125 psig LPCI pump discharge pressure represents a pump flow of 5200 gpm.

The value obtained from using the nominal LPCI pump flow value of 5000 gpm (resulting in a discharge pressure at the outlet of the heat exchanger of 141.1 psig), 10 psig for the suppression pool over pressure post LOCA and 14.0 psig calculated for CCSW line losses (at nominal flow of 3500 gpm) to the outlet of the heat exchanger results in an available dP across the heat exchanger of 14.9 psid which is greater than the 8.26 psig required and therefore demonstrates adequate protection against effluent discharge.

Memorandum of Conversation **SARGENT & LUNDY**

	Date 02-19-93 Time 9:00 a.m.
Person J. W. Dingler	Company S&L
Person R. J. Goebbert	Company S&L
Project Dresden	Project No. 9188-10

Subject Discussed 20 psid Differential Pressure Setpoint Between CCSW and LPCI Across LPCI HX

Summary of Discussion, Decisions and Commitments

1. Mr. Goebbert contacted Mr. Rob Davison at GE (408-925-4587). Mr. Davison is the DBD writer for RHR. He provided the following information.

2. The design basis for the pressure differential is that the CCSW always be positive in respect to LPCI.

3. The 20 psid value was set based on consideration of instrument and control error plus 5 psi.

4. This design bases is similar for other GE plants. It will be documented in a GE internal correspondence which will be provided to CECo with the LPCI/RHR DBD documents.

cc J. Dawn
S. Eldridge
G. Lupia
T. R. Eisenbart
D. Bianchini
B. Barth
T. Behringer
R. J. Goebbert
L. Schwarz

File

J W Dingler
Signature