Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

March 18, 1993

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Commonwealth Edison's (CECo) Review of the NRC Systematic Evaluation Program (SEP) Concerns Regarding CECo's Validation of the Design Basis of the Containment Heat Removal System
NRC Docket Nos. 50-237 and 50-249

References:
(a) Enforcement Conference held between members of CECo and the NRC Staff to discuss matters related to the CCSW System at Dresden Station, dated February 22, 1993.

(b) P. Piet letter to T. Marl, dated March 5, 1993.

(c) D. Crutchfield (U.S. NRC) letter to L. DelGeorge (CECo), dated December 28, 1981.

(d) T. Rausch (CECo) letter to P. O'Connor (U.S. NRC), dated April 6, 1982.

(e) P. O'Connor (U.S. NRC) letter to L. DelGeorge (CECo), dated August 19, 1982.

Dr. Murley:

As discussed during the Reference (a) meeting, CECo has reviewed the pertinent sections of the SEP to determine the impact of postulating reduced LPCI heat exchanger capability on the overall conclusions of the SEP evaluations. CECo's review centered on the information contained in References (c), (d), and (e). CECo's evaluation has been completed and is provided as an attachment to this letter.

a/plp5/nfssep/1

250039

A001 /1

The original focus of the NRC SEP evaluations as it pertains to the subject issue , was primarily on peak containment pressure for structural considerations and on drywell temperature for Equipment Qualification purposes. The new analyses performed (Reference (b)) to validate the design basis of the Containment Heat Removal System (CHRS) for long-term containment cooling using one LPCI pump and one CCSW pump scenarios adequately address long-term suppression pool temperature concerns. Therefore, no re-analysis of the SEP cases is warranted.

The impact of the changes to LPCI/CCSW system assumptions have been analyzed by CECo, focusing on long-term suppression pool behavior, and shown to yield acceptable results for the large break LOCA case. This case bounds all other break sizes with respect to energy deposition to the suppression pool. The drywell temperatures and pressure peaks for the steam break cases occur prior to or in close proximity to the initiation of pool cooling, and therefore would not be significantly affected by changes to LPCI/CCSW assumptions.

As discussed in the attachment to this letter, CECo has re-analyzed the 0.01 ft2 steam break to determine if the impact of changes in LPCI/CCSW system assumptions on Mark I containment local pool temperature limits. This analysis used conservative inputs and assumptions and included the actions required by the Dresden Technical Specifications to depressurize the reactor upon reaching high suppression pool temperatures. Margins to local pool temperature limits have been demonstrated.

In conclusion, CECo has identified the potential impacts of LPCI/CCSW assumptions on the NRC SEP evaluation. The principal conclusions of the SEP are unaffected. CECo has performed analysis that demonstrate that the results using the alternate LPCI/CCSW assumptions (1 LPCI/1 CCSW, revised HX data sheet) are acceptable.

If there are any questions concerning this matter, please contact this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachment: An Evaluation of LPCI Heat Exchanger Flow Parameter Changes on the NRC SEP Containment Analysis for Dresden Station

cc: A.B. Davis, Regional Administrator - RIII
J.F. Stang, Project Manager - NRR
P. Hiland, RIII
M.N. Leach, Senior Resident Inspector - Dresden
Office of Nuclear Facility Safety

An Evaluation of LPCI Heat Exchanger Flow Parameter Changes on the NRC SEP Containment Analysis for Dresden Station

Introduction

Recent reevaluations of the CCSW and LPCI systems as applied to post-LOCA containment cooling functions were based on the determination that the Dresden design basis limiting containment cooling case is with one LPCI and one CCSW pump. Reanalysis of the post-LOCA suppression pool heatup showed that acceptable results were achieved with this cooling assumption. It was noted that the FSAR needed clarification and updates were prepared.

NRC review of this effort resulted in questions being raised relative to the assumptions, methods, and conclusions of the Systematic Evaluation Program (SEP) evaluation report on Topics VI-2.D and VI-3 ("Containment Pressure and Heat Removal Capability/Mass Energy Release for Possible Pipe Break Inside Containment") [Ref. 1] It was noted that this effort was based in part on FSAR information, and that the impact of changes should be addressed.

The purpose of this evaluation is to review the inputs to, methods applications, and conclusions of the SEP analysis and discuss/disposition the differences.

Description of SEP Evaluation

The SEP Evaluation was performed by Lawrence Livermore Labs using a combination of FSAR information and licensee supplied data. The intent of the analyses was to demonstrate that Dresden Station, when analyzed using Standard Review Plan based methods and assumptions, would show adequate safety margins, particularly with respect to peak containment pressure and temperature. The conclusions of the SEP evaluation were:

> "The confirmatory analysis was found to be comparable to the licensee's results and, therefore, we conclude that the licensee has satisfactorily demonstrated the adequacy of the containment functional design. The licensee has not submitted a steam line break analysis, therefore, an independent analysis was performed...The containment atmosphere conditions during a 1.0 sq. ft. steam line break may be utilized as input to the equipment qualification of safety related equipment effort." [Ref. 1]

As noted above, the SEP evaluation performed a series of confirmatory containment response analyses. These were performed using the CONTEMPT-LT-028 and the RELAP4 Mod 007 computer codes. A RELAP model, developed based on RETRAN input data provided by CECo was used to generate the mass and energy release information. The CONTEMPT model was principally based on FSAR available data. The cases that were run are described below:

1. Recirculation Line Pipe Breaks - a 5.62 ft2 break was assumed, and blowdown was calculated out to the time at which the vessel pressure dropped below the drywell design pressure (62 psig). Subsequent to end of blowdown, decay energy comparable to 1.2 * ANS curves was used as input to the vessel model in CONTEMPT. Containment cooling was stated to be based on two LPCI pumps in operation, manually initiated at 600 seconds into the event.

2. Main Steam Line Breaks - Two cases were analyzed (1.0 FT2 and .01 FT2). As in the previous cases, the M/E releases were calculated with RELAP and blowdown tables were developed to the time at which the reactor pressure equaled the containment design pressure. Decay heat loads were then assumed beyond this point.

The containment data used in this evaluation is presented in Table 1.

Description of CECo Analysis

The purpose of the CECo reanalysis efforts was to reconstitute a design basis analysis consistent with the regenerated LPCI heat exchanger performance data developed in early 1992. This effort was initiated as a result of plant questions regarding the design flow rates of the LPCI heat exchangers. Since original design calculations on the LPCI heat exchangers could not be completely recovered, a decision was taken to regenerate this information. Discrepancies between the original FSAR stated LPCI HX performance and the regenerated calculations were identified, with the new heat transfer performance conservatively predicted to be approximately 9% less. Therefore, a long term post-LOCA containment thermal analysis was performed to demonstrate adequacy [Ref. 2]. This analysis was based on a large break scenario. Key input assumptions and data used in this analysis is presented in Table 2.

The key results of interest in this reanalysis were the long term suppression pool temperature and pressure. These values were then used as a basis in ECCS pump NPSH calculations, and to determine the ability of the CCSW pumps to maintain a positive pressure differential to the LPCI system, precluding radionuclide intrusion into the service water systems.

An additional analysis was performed to demonstrate that the revised LCPI HX performance would not invalidate condensation stability transients performed in support of the Mark I containment program [Ref. 3]. The limiting transient was a small steam line break (.01 FT2) on the isolation condenser steam line. The results of this calculation demonstrated that margin exists to local suppression pool temperature limits for discharge of steam through the S/RV T-Quenchers. Key assumptions and inputs used in this analysis are tabulated in Table 3.

Table 1 Key Analysis Assumptions, SEP Evaluation

Initial Conditions	
Reactor Power	102%
Reactor Dome Pressure	1020 psia
Suppression Pool Initial Temperature	125 F
Pool Atmosphere Initial Conditions	125 F, 100% RH, 0.0 psig
Drywell Free Volume	158236 ft3
Suppression Pool Airspace Volume	117245 ft3
Suppression Pool Volume	112203 ft3
Time to initiate pool cooling	600 seconds
Drywell initial conditions	150 F, 100% RH, 0.0 psig
Downcomer submergence/flow area	3.417 ft, 285 ft2
Feedwater mass/energy	not included in large break, constant h until trip for MSLB cases
Decay Heat	1.2 x ANS 1971
ECCS pump heat	not included

Heat Exchanger Capability

	(2 LPCI/2 CCSW)
Heat Removal (at reference temperatures below)	102 MBTU/HR
Primary Flow (LPCI)	10700 GPM
Secondary Flow (CCSW)	7000 GPM
CCSW Temperature	95 F
LPCI Temperature	165 F

Table 2 Key Analysis Assumptions, CECo/GE Recirc Line Break

Initial Conditions	
Reactor Power	102%
Reactor Dome Pressure	1020 psia
Suppression Pool Initial Temperature	95 F
Pool Atmosphere Initial Conditions	95 F, 100% RH, 0.15 psig
Drywell Free Volume	158236 ft3
Suppression Pool Airspace Volume	120097 ft3
Suppression Pool Volume	112000 ft3
Time to initiate pool cooling	600 seconds
Drywell initial conditions	135 F, 20% RH, 1.25 psig
Downcomer submergence/flow area	3.67 ft, 301.6 ft2
Feedwater mass/energy	feedwater mass and energy added to containment
Decay Heat	ANS 1979
ECCS pump heat	100% of motor HP added as heat input

Revised Heat Exchanger Capability with maximum flow uncertainties

	(1 LPCI/1 CCSW)	(2 LPCI/2 CCSW)
Heat Removal (at reference temperatures below)	55.238 MBTU/HR	82.48 MBTU/HR
Primary Flow (LPCI)	3881 GPM	8916 GPM
Secondary Flow (CCSW)	3071 GPM	4795 GPM
CCSW Temperature	95 F	95 F
LPCI Temperature	165 F	165 F

Table 3 Key Analysis Assumptions, CECo 0.01FT2 Steamline Break Transient Analysis

Initial Conditions	
Reactor Power	102%
Reactor Pressure	1050 psia
Pool Initial Temperature	101 F (1F Steady State Error)
Suppression Pool Temperature Monitoring System Error	6F due to all causes (Transient Error)
Steam flow feed flow	102% of licensed conditions
CRD flow	11.11 lb/sec
Offsite power available for all cases	
Normal Auto operation of ECCS systems	
MSIV Closure Time	3 seconds with .5 sec delay
Feedwater temperature	operating condition until feed system volume is swept, then 170 F used for conservatism
Decay Heat	Approximately 1.1 ANS 1979 for infinite operating history
LPCI pumps heat	100% of motor HP added to pool as heat input

Revised Heat Exchanger Capability

	(1 LPCI/1 CCSW)	(2 LPCI/2 CCSW)
Heat Removal (at reference temperatures below)	63.63 MBTU/HR	89.06 MBTU/HR
Primary Flow (LPCI)	4500 GPM	9000 GPM
Secondary Flow (CCSW)	3450 GPM	5000 GPM
CCSW Temperature	95 F	95 F
LPCI Temperature	165 F	165 F

Impact of LPCI HX Changes on SEP Evaluations

The principal impact of the LPCI HX changes on the SEP evaluation is in the value of containment cooling capability assumed. As noted previously, the SEP evaluation appears to have used a value of 102 million BTU/HR at the heat exchanger design table temperature conditions (165 F Suppression Pool, 95 F Service Water). This would require 2 LPCI pumps and 2 CCSW pumps to achieve, as defined in the original design. Using the assumptions applied in the CECo analysis, the heat removal capability would be based on a 1 LPCI/1 CCSW pump combination. A comparison of the flows and heat transfer rates is shown in the previous tables for various pump combinations.

[Note: Slightly different heat removal rates are shown for the CECo large break vs small break. The large break values are based on reconstituted LPCI HX performance at the minimum flows (maximum uncertainties). The small break values are based on the original HX data sheet values and use NTU methods to obtain the low flow values. The small break analysis was rerun using the reconstituted HX performance to ensure consistency. The original values used are included here to demonstrate the discrepancies in HX performance that led to the reanalysis effort.]

An effort has been made at CECo to reconstruct the containment analyses performed in the SEP evaluation to facilitate understanding of the impact of reduced heat exchanger performance on the conclusions of the SEP study. In general, the containment response to the blowdown portion of the transients can be reproduced virtually exactly. The subsequent decay heat removal results of the SEP analyses exhibit behaviors that are not readily explainable, and may be due more to the methods applied in the analysis than to the physical realities of the problem analyzed. The following examples provide further explanation of this point:

1. On the 5.62 FT2 Recirculation Line Break, the wetwell temperature rises quickly for the first 50 seconds, after which it is completely flat. At 600 seconds, following the initiation of pool cooling, the pool temperature declines. This would imply that no energy addition to the containment occurred during this interval.

2. The decay power table is referenced as being 1.2 times the ANS 5.1 (1971) curve. A check of the values utilized confirms that this is the case, with the exception of the 40000 second data point, which appears to be in fact 1.2 times the decay power at 400000. It is also noted that the SRP position on decay heat contained in ASB 9-2 would require a factor of 1.1 times the ANS 5.1 (1971) after 1000 seconds.

3. The containment temperature, particularly the wetwell temperature, should continue to rise due to the addition of decay heat until the heat exchangers balance the heat addition. Figure 1 includes a plot of the stated decay energy used compared to the heat exchanger performance at the FSAR design point referenced by the analysis (165 F suppression pool). As can be seen, the balance does not occur between containment heat removal and decay heat addition until approximately 10000 seconds (if the suppression pool is at least 165 F and the service water is 95 F, delta T=70 F). Based on review of the analyses, the turnaround point

occurs at 600 seconds for the large break with a delta T of 73 F, 11000 seconds for the small steam break with a delta T of 43 F, and 3000 seconds with a delta-T of 66 F for the 1.0 FT2 steam break. It is noted that earlier turnover would be possible if a second LPCI HX were assumed operable, however, this would be inconsistent with the single failure of a diesel generator assumed in the evaluation.

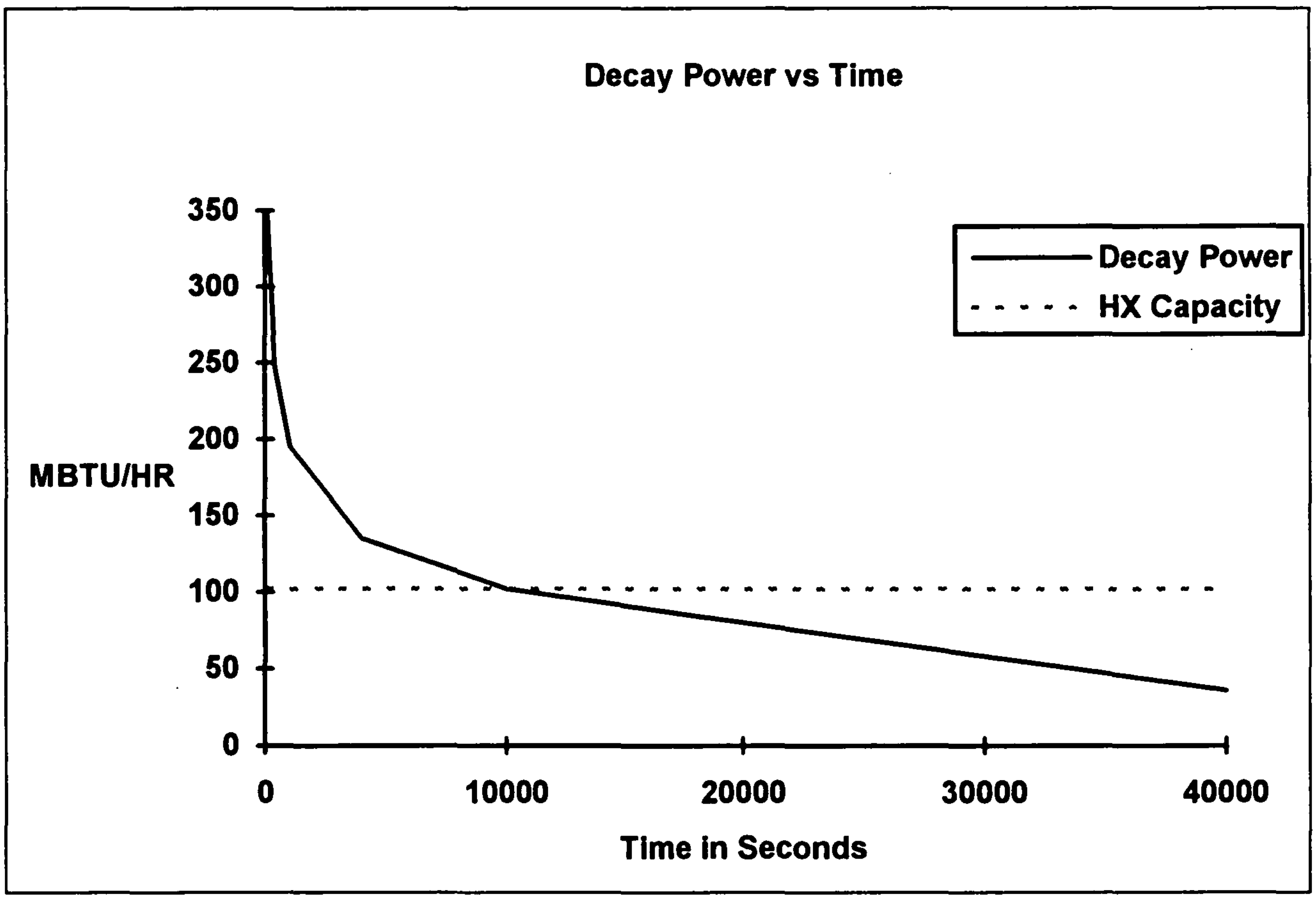
It is believed that the discrepancies cited above are most likely related to the method of implementing the primary system model in the CONTEMPT code model. The analyst is required to input the primary system internal energy and mass at the end of blowdown, which are then used in conjunction with the decay energy table inputs. The code then determines the amount of primary fluid which would boiloff to the drywell vapor region, or spill as liquid to the drywell floor. Errors in the initial energy or mass, or in subsequent additions to the primary system can have profound effects on the post-blowdown containment calculation.

In fairness to the analysts (and reviewers), it should be pointed out that the principal results of interest were the peak drywell pressures and temperatures. Long term containment cooling was not of particular concern, and was not highlighted in any of the SEP conclusion statements. The response to a CECo comment also indicates that the focus of the analysts was on peak pressure, and not on long term containment temperature:

> "The staff agrees that the initial wetwell temperature of 125 F is 30 F greater than the Technical Specifications allow. The effect of this would be to slightly raise the drywell and wetwell pressures. Since, as previously mentioned, these are well within allowable values the staff does not believe this to be a significant factor in affecting the conclusions reached concerning the containment pressure response." [Ref. 1]

Figure 1

Decay Power vs Time
LPCI HX Design Point Performance



Conclusions

As stated above, CECo believes the predominant concern of the personnel performing the SEP evaluation was the peak drywell pressure and temperature. The changes to the CCSW/LPCI systems assumptions would have no impact on the peak drywell pressures and temperatures, and the SEP conclusions remain valid.

The impact of the changes to CCSW/LPCI system assumptions have been analyzed by CECo in analysis specifically focused on long term suppression pool behavior, and shown to yield acceptable results for the large break LOCA case. This case bounds all other break sizes with respect to energy deposition to the suppression pool. The drywell temperatures and pressure peaks for the steam break cases occur prior to or in close proximity to the initiation of pool cooling, and therefore would not be significantly affected by changes to CCSW/LPCI system assumptions either.

CECo has also reanalyzed the 0.01 FT2 steam break to determine the impact of changes in CCSW/LPCI system assumptions on Mark I containment local pool temperature limits. This analysis used conservative inputs and assumptions and included the actions required by Technical Specifications to depressurize the reactor upon reaching high suppression pool temperatures. Margins to local pool temperature limits have been demonstrated.

In conclusion, the potential impacts of CCSW/LPCI assumptions on the SEP evaluation have been identified. The principal conclusions of the SEP are considered to be unaffected. CECo has performed analysis that demonstrate that the results of using the alternate CCSW/LPCI assumptions [1LPCI/1CCSW combination, revised HX data sheet] are acceptable.

References

1. "Systematic Evaluation Program (SEP) for Dresden Nuclear Power Station, Unit 2 - Evaluation Report on Topics VI-2.D and VI-3", letter Paul O'Connor to L. DelGeorge with attachments dated August 19, 1982.

2. "GENE-770-26-1092", General Electric Containment Reanalysis with reduced CCSW/LPCI system performance dated October 1992.

3. "An Evaluation of Reduced CCSW Flows at Dresden Station", RSA-D-92-01 dated April 1992.