Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

August 14, 1992
LIC-92-285R

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

SUBJECT: Transmittal of Reactor Vessel Integrity Calculation Summary for Fort Calhoun Station

On July 3, 1992, Omaha Public Power District (OPPD) experienced a small break loss of coolant event at Fort Calhoun Station Unit No. 1. OPPD has performed an analysis of the reactor vessel which verifies that no adverse effects resulted from this event. During the week of July 27, 1992, Dr. J. T. Larkins of the NRC requested that OPPD submit the attached analysis titled "Fort Calhoun Station Reactor Vessel Integrity Calculation Summary."

If you should have any questions, please contact me.

Sincerely,

J K Gasper

for W. G. Gates
Division Manager
Nuclear Operations

WGG/sel

Attachment

c: LeBoeuf, Lamb, Leiby & MacRae
J. L. Milhoan, NRC Regional Administrator, Region IV
R. P. Mullikin, NRC Senior Resident Inspector
S. D. Bloom, NRC Acting Project Manager
J. T. Larkins, NRC Director, Project Directorate IV-1

A001
1/1

9208190005 920814
PDR ADOCK 05000285
S PDR

Fort Calhoun Station Reactor Vessel Integrity Calculation Summary

August 14, 1992

With respect to the July 3, 1992, small break loss of coolant event at Fort Calhoun Station (FCS), Omaha Public Power District (OPPD) has performed an evaluation related to the establishment of natural circulation and a conservative analysis for stagnant reactor vessel downcomer flow utilizing the REMIX code.

Based upon computer simulations of natural circulation events at other Combustion Engineering (CE) designed plants, flow into the downcomer decreased to about 3% of full flow at its lowest point following the trip of the final two reactor coolant pumps (at 23:49 on July 3, 1992). The ability of FCS to transition into natural circulation following a pump trip is one of the fundamental design bases of the CE reactor coolant system (RCS). To stagnate flow and prevent the plant from going into natural circulation during the time frame immediately following a pump trip is considered improbable.

Subcooled natural circulation is verified by the operator per the appropriate floating steps in Emergency Operating Procedure EOP-20, Functional Recovery Procedure, as indicated by: (1) ΔT (cold leg and hot leg) is less than or equal to 50°F, (2) the difference between core exit thermocouples (CET) and RCS T_{hot} is less than or equal to 10°F, (3) RCS T_{hot} and T_{cold} temperatures are stable or lowering and (4) RCS subcooling is greater than or equal to 20°F. By confirming that all of the above requirements are met, the operator is assured that natural circulation is present in the RCS.

The operator logged confirmation of natural circulation during the transient at 00:04 on July 4, 1992. Natural circulation most likely existed earlier due to the fixed geometry of the RCS, i.e., the position of the core relative to the steam generator, and the inventory of relatively cold water that exists above the core in the cold legs and in the steam generator U-tubes. However, some time is required for development of a stable core ΔT and subsequent verification of the natural circulation criteria by the operator using EOP-20. Natural circulation provides sufficient mixing to preclude flow stratification in the cold leg/downcomer regions. Based on data from the event, OPPD and CE qualitatively concluded that natural circulation was established. However, since a detailed computer simulation for the July 3 event is not available, a conservative assumption of flow stagnation was made to ensure a bounding lower temperature was obtained.

CE performed an evaluation of the thermal-hydraulic transients resulting from the injection of high pressure safety injection (HPSI) flow into a RCS cold leg during a period of postulated flow stagnation in the RCS loop. The resulting thermal stratification in the cold leg and the vessel downcomer is of importance for pressurized thermal shock (PTS) related evaluations. The FCS safety injection (SI) nozzle is at a 75° angle, relative to the center line of the cold leg, for injection into the cold leg which enhances mixing with the RCS fluid.

Background Information on REMIX:

In general, transient system codes such as RELAP or RETRAN provide bulk coolant temperatures and not local temperature distributions which may result from thermal stratification phenomena. These codes assume uniform mixing in each node. This assumption is valid as long as forced or natural circulation is present in the loop and downcomer. Typically, natural circulation flows are several times the HPSI flows, which would result in good mixing of the loop flow with the HPSI flow. Flow stratification would be obtained only during flow stagnation.

The regional mixing model of the REMIX code calculates the effect of HPSI flow stratification in the cold leg and the downcomer during loop flow stagnation. The models of the REMIX code have been successfully used to interpret test data (Reference 2) within the parameters experienced at FCS. The use of REMIX at higher system pressure (1000-1250 psi) and lower HPSI flow rates (200 gpm total) experienced during the FCS transient is within the conservative bounds of validity of the REMIX code.

Physical Model of REMIX Code:

The physical situation, modelled under assumed stagnation conditions, is depicted in Figure 1. The relevant portions of the RCS which participate in the regional mixing include all fluid particles that can reach the injection nozzle through a sequence of horizontal and upward vertical translation. These include the loop-seal, cold leg, downcomer and the lower plenum volumes.

Safety injection flow enters the system through the SI line and an equivalent flow rate exits through the reactor core, as dictated by continuity considerations. For all practical SI rates, a stratified cold leg configuration is obtained as shown in Figure 1.

A "cold stream" originates with the injected stream, continues towards both ends of the cold leg and decays away as the resulting plumes fall into the downcomer and pump/loop-seal regions. A "hot stream" flows counter to the "cold stream" supplying the flow necessary for mixing (entrainment) at the mixing regions shown in Figure 1 (MR-1 to MR-5). Significant mixing of the HPSI flow with the "hot stream" occurs right at the point of injection (MR-1). In MR-3 and MR-5 mixing occurs because of transitions from horizontal layers into falling plumes. Negligible mixing occurs in MR-2, the interface between the hot and the cold streams in the cold leg. MR-4 is the region where the downcomer plume finally decays.

From a practical standpoint the minimum (centerline) downcomer plume temperature is of significance. It is governed by the strength of the downcomer plume which, in turn, is dictated by the extent of the cold leg stratification.

Event Specific Input for the REMIX Code

A period of no flow conditions in the cold legs and downcomer was conservatively assumed to have taken place immediately following the end of coastdown of the second pair of reactor coolant pumps, RC-3A and RC-3C (23:55 on July 3, 1992). This period was assumed to last until natural circulation flow was operationally confirmed at 00:04 on July 4, 1992.

Review of plant data from the Qualified Safety Parameter Display System (QSPDS) with respect to cold leg temperature transients, reactor coolant pump (RCP) operation and safety injection cycles indicated that cold leg 1A and the reactor vessel downcomer region below the reactor vessel inlet nozzle 1A are the most limiting with respect to the coolant temperature. Assuming the lowest initial coolant temperature results in a conservative final fluid temperature in the downcomer.

Accordingly, Cold Leg 1A temperature (QSPDS channel TA112C) data was used to identify the appropriate initial coolant temperature as input to the REMIX code. The identified temperature value, 494°F, is the lowest cold leg temperature just prior to and during the assumed stagnation period. For most of this period, however, the coolant temperature was around 520°F.

Summary of REMIX Results:

The transient was simulated using the REMIX computer code (Reference 1). The results for the cases are summarized in Table 1. All cases conservatively assume nine minutes of flow stagnation and a constant average safety injection (SI) flow rate into the stagnated cold leg, even though actual event data quantitatively indicates the establishment of natural circulation. The initial loop temperature utilized is 494°F; the SI flow temperature utilized is 80°F, the safety injection and refueling water tank temperature noted in the control room log.

Case 1 utilizes the lower plenum volume as a part of the total mixing volume in the REMIX model. For additional conservatism this mixing volume was modified for Case 2. Case 2 assumes the lower plenum volume does not participate in the regional mixing process. This assumption is conservative and results in a lower downcomer plume temperature.

Table 1 lists the calculated downcomer plume centerline temperatures. The plume centerline temperature is at 10.4 feet below the cold leg centerline which is at the top of the most limiting 3-410 reactor vessel (longitudinal) weld. Other key RCS parameters from the event are also shown in Table 1.

ASME Section XI Appendix E:

Appendix E of the ASME Code Section XI provides acceptance criteria and guidance for performing an engineering evaluation of the effects of an out-of-limit condition on the structural integrity of the reactor vessel beltline region. Showing compliance with either Paragraphs E-1200 or E-1300 assures that the beltline region has adequate structural integrity for the unit to return to service.

For thermal transients where $\Delta T_c/\Delta t \geq 10°F/hr$, Paragraph E-1200 states adequate structural integrity of the reactor vessel beltline region is assured if the following criteria are satisfied throughout the event:

1) $T_c - RT_{NDT} > 55°F$ and
2) Maximum pressure does not exceed design pressure.

The minimum temperature conservatively calculated for T_c in the beltline region is 363°F as shown in Table 1. The limiting RT_{NDT} value used in this analysis is 242°F for the 3-410 weld. Therefore, $T_c-RT_{NDT}=121°F$, which is much greater than 55°F. The maximum calculated pressure for the transient, including uncertainties, was 2472 psi which was less than the design pressure of 2500 psi. Thus, it was safe for the reactor vessel to return to and be in service.

To provide an additional measure of assurance that operation of the FCS reactor vessel is safe, the more rigorous approach of evaluating the vessel condition as specified in paragraph E-1300 was evaluated. The coolant temperature, calculated by REMIX, is input in the form of a curve describing the temperature response of the coolant at the wetted surface of the beltline region during the event. The temperature response is used in a heat transfer analysis to provide a detailed temperature profile of the metal temperature through the vessel wall for all time points in the event. The temperature profile is then used to calculate K_{It} for the time points. The K_{It} result is combined with other mechanical loads, K_{Im} and K_{Ir}, which are also calculated in this process. This sum is compared to K_{Ic}, in the below listed relationship, to establish the margin to crack initiation throughout the transient.

$$1.4(K_{Im} + K_{It}) + K_{Ir} \leq K_{Ic}$$

If the equation is satisfied for all time points, the criterion is met and the vessel is safe. The minimum temperature value corresponding to the plume centerline temperature and satisfying this equation is 135°F. Therefore, since 363°F is much greater that 135°F, the requirements of paragraph E-1300 are also met.

Additional conservatism was also incorporated into the evaluation. The heat transfer coefficient used to determine the temperature distribution through the vessel wall was assumed to be 1000 BTU/hr ft^2F. A more typical value for natural circulation flow would be 300 BTU/hr ft^2F. The temperature profile used the lowest temperature on the centerline of the plume on the entire vessel; the crack depth used in the analysis is 28% greater than the crack depth required to be used in Appendix E.

The results from the recently completed March 1992 vessel 100% inservice inspection show that an indication even close to the 1.00 inch crack depth required by Appendix E does not exist in the FCS vessel. A smaller crack would increase the margin in E-1300.

References:

1. K. Iyer, H.P. Nourbaksh, T.G. Theofanous, "REMIX: A Computer Program for Temperature Transient Due to High Pressure Injection After Interruption of Natural Circulation", NUREG/CR-3701 R2, May 1986.

2. T.G. Theofnaous, and H. Yan, "A Unified Interpretation of One-Fifth to Full Scale Thermal Mixing Experiments Related to Pressurized Thermal Shock", NUREG/CR-5677, April 1991.

TABLE 1

Calculated Temperatures Nine Minutes After Loop Stagnation

Case No.	Lower Plenum In Model	SI Flow Rate, GPM	Plume Center Line Temp*,°F
1	Yes	100	386
2	No	100	363

* This is the downcomer plume centerline temperature at 10.4 Ft. below the cold leg centerline which is the top of the limiting longitudinal seam weld 3-410.

The results (temperatures) listed above are for the analysis assumption of nine minutes of stagnation in the loop and are based on the following data:

Initial loop temperature = 494°F

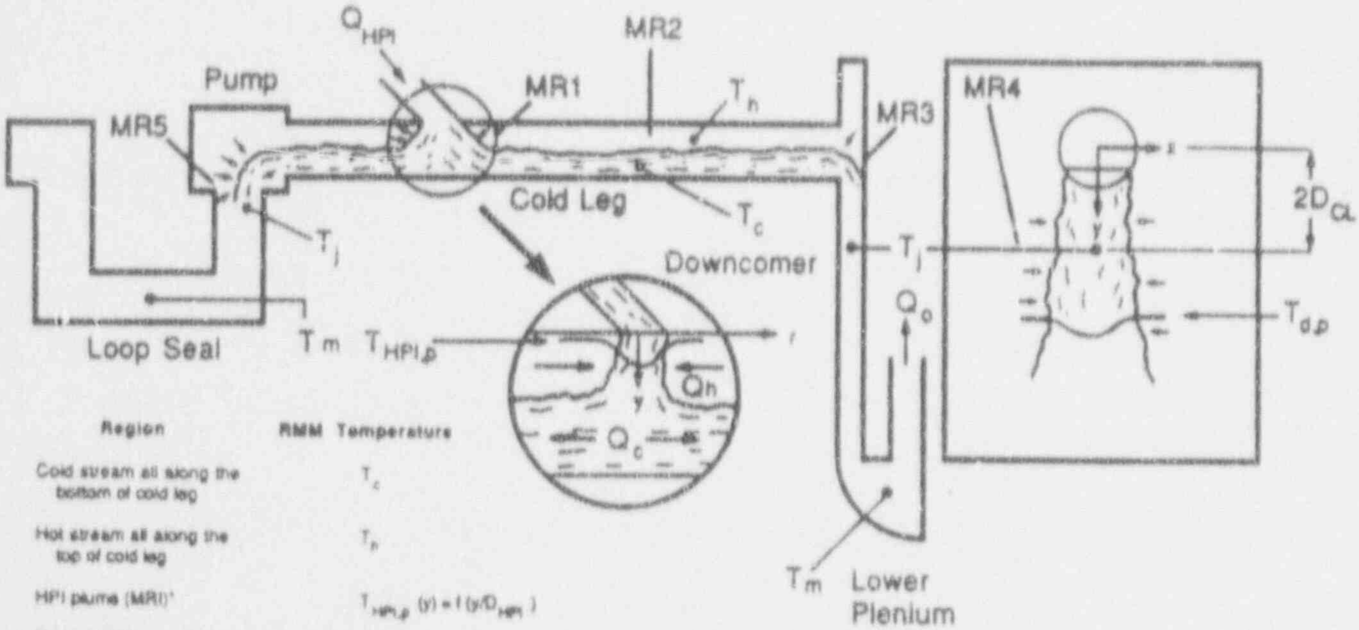
SI flow temperature = 80°F

System pressure = 1000 psia

SI injection in two loops only

Figure 1

The Temperature Zones of the Regional Mixing Model (RMM)



Region	RMM Temperature
Cold stream all along the bottom of cold leg	T_c
Hot stream all along the top of cold leg	T_h
HPI plume (MRI)*	$T_{HPI,p}(y) = f(y/D_{HPI})$
Pump (MR5)	T_j
Loop Seal	T_m
Downcomer plume at MR3	T_j
Downcomer plume (MR4)**	$T_{d,p}(0,y) = g(y/D_{CL})$
Outside of downcomer plume	T_m
Lower plenum	T_m

* Mixed mean temperature is computed using entrainment as a function of y and Fr_{HPI}

**This temperature is the lowest for the given level.