Omaha Public Power District

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April 30, 1982
LIC-82-176

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

The Commission's letter dated March 18, 1982 requested the Omaha Public Power District provide additional information regarding the pressurized thermal shock issue at the Fort Calhoun Station. The District's response is attached.

In addition to the specific responses, materials data were developed. One area of conservatism included in all evaluations reported to date is the use of the ASME Section XI method for predicting K_{IC} for a given value of RT_{NDT}. The specific material test samples included in the data used to develop the ASME XI K_{IC} curves were reviewed. It was found that certain materials which are not representative of Combustion Engineering (CE) reactor vessel materials were included in the development of the curves. Restriction of the data to only materials representative of CE reactor vessels would result in an effective increase in K_{IC} equivalent to an RT_{NDT} reduction of about $30^{\circ}F$. This is a significant amount of conservatism which should ultimately be factored into the assessment of the pressurized thermal shock situation.

As part of the ongoing pressurized thermal shock program, the CE mixing model used in the CEN-189 report was evaluated against the more recent EPRI/Creare scale model mixing tests. These recent tests evaluated mixing of cold HPSI water in a system with top injection into a horizontal cold leg pipe. The CE methodology was shown to conservatively underpredict the excellent mixing observed in the tests. Both the tests and the CE computations indicate that forward loop flow produces essentially complete mixing within the cold leg piping prior to entering the reactor vessel annulus.

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Based on the evaluations previously submitted and considering the additional information provided here, the District continues to believe there is no near term pressurized thermal shock problem at the Fort Calhoun Station and, therefore, an orderly program for resolution of this concern is proper.

Sincerely,

W. C. Jones
Division Manager
Production Operations

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

OMAHA PUBLIC POWER DISTRICT'S
RESPONSE REGARDING PRESSURIZED THERMAL SHOCK

Concerning Operator Actions

1. In CEN-189, only two cases are considered for a SBLOCA with concurrent loss of feedwater. In one case, PORVs are opened by the operator at 10 minutes to prevent core uncovery. In the other case, feedwater is restored to the steam generator in 30 minutes to prevent core uncovery. For both cases, the report stated that 15-30 minutes would provide ample time to initiate feedwater prior to dryout. Provide the analysis or basis to justify that 15 to 30 minutes is ample time for correct operator action.

2. In CEN-189, provide an evaluation of the sensitivity of the transient to the time assumed for operator action (i.e., if the operator opens the PORVs at 15 minutes, or 30 minutes, or restores feedwater alone at 15 minutes, or 20 minutes, or 45 minutes, what are the resulting pressure/temperature transients?).

Response

CEN-189 was submitted in response to Action Item II.K.2.13 of NUREG-0737 which required that a detailed analysis be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. The recommended mode of decay heat removal for the CE NSSS is by means of the steam generator secondary side system. Therefore, loss of all feedwater as required by II.K.2.13 represents a potential inadequate core cooling situation. Since the concern in II.K.2.13 is the thermal-mechanical conditions of the vessel, only scenarios which permit adequate core cooling were considered, in order that "recovery" could be assured. Accordingly, CE reviewed CEN-114, "Small Break Transients in CE NSSS's", which was submitted to the Commission in 1979, and chose two different basic modes of "recovery" from loss of all feedwater.

One mode of recovery selected from CEN-114 for pressurized thermal shock evaluation was to reestablish auxiliary feedwater. The CE U-tube steam generators have sufficient secondary side inventory, such that 15 to 30 minutes is required to dry out a steam generator after loss of all feedwater. Reestablishing feedwater prior to dryout would represent a less severe PTS situation due to mixing of the auxiliary feedwater with the remaining contents of the steam generators. The 30 minutes chosen for reestablishing feedwater, as reported in CEN-189, represents the time after which the steam generators would be essentially dry and this case was basically an over-feed transient to dry steam generators. Therefore, the results of the PTS analysis would not be affected by reestablishing feedwater after 30 minutes because steam generators are essentially dry at 30 minutes.

The other mode of recovery selected for PTS evaluation was to open PORV's to prevent inadequate core cooling, assuming a total and continued loss of all feedwater. This method of cooldown is not advocated by CE, but was evaluated simply to satisfy the requirement of II.K.2.13 to assume loss of all feedwater. The operator action of opening the PORV's at 10 minutes is completely dictated by the core cooling aspects of this scenario and is not subject to variation for PTS considerations.

Since the parameters for the two types of cases reported in CEN-189 were chosen to maximize the PTS aspects of the transients (within the limits of maintaining adequate core cooling) and since these cases were found to be less governing than the cases reported in the letters responding to the 150-day request of the NRC's August 21, 1981 letter, we believe sufficient variation of the CEN-189 cases has been accomplished.

3. The Ft. Calhoun analysis took credit for warm prestressing, but stated that to preserve warm prestressing (in some cases), operator action is necessary to maintain plant parameters within pressure temperature limits. Provide an evaluation of the probability of operator error for all cases where credit is taken for warm prestressing based on operator action.

Response

The Fort Calhoun Station analysis took credit for warm prestressing in Case 4 (Table 4-4) reported in CEN-189. Warm prestressing was not credited in any of the other analyses.

CEN-189 contains analyses related to NUREG-0737, Item II.K.2.13. Action Item II.K.2.13 requires that "a detailed analysis shall be performed on the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater". The requirement "deals with the potential for thermal shock of reactor vessels resulting from cold safety injection flow". "In particular, demonstration shall be provided that sufficient mixing would occur of the cold high pressure injection water with reactor coolant so that significant thermal shock effects to the vessel are precluded."

Two types of SBLOCA - LOFW scenarios, with four different cases of each type, were evaluated in the subject report. The first type assumed loss of all feedwater, including auxiliary feedwater, and that the PORVs are used to provide a means of decay heat removal. The second type of SBLOCA - LOFW scenario assumed:

1) recovery of auxiliary feedwater after loss of feedwater has persisted long enough to allow dryout of both steam generators, and

2) refill of the generators at maximum auxiliary feed pump capacity in order to conservatively maximize the cooldown effect.

Case 4 analysis, as reported in CEN-189, assumed warm prestressing and is the limiting case for the first type of SBLOCA - LOFW scenario analyzed.

The attached Figures 1 and 2 demonstrate a slow, steady decrease of RCS pressure and temperature after the operator opens the second PORV at 10 minutes into the transient. Failure of the operator to open the second PORV at or before 10 minutes would require restoration of auxiliary feedwater at 30 minutes into the transient to maintain adequate core cooling. Case 5 in CEN-189 is the limiting case for restoration of feedwater and shows acceptable results without credit for warm prestressing. Case 5 is a more severe PTS transient compared to Case 4 without opening the PORV and restoring feedwater. Figure 3 shows warm prestressing is only credited for Case 4 after 60 minutes of the transient have elapsed. Operator actions at this time would be to continue to cool and depressurize the RCS using the HPI system and PORVs and to restore feedwater to one steam generator. If feedwater can be restored, the operator would then lower the RCS pressure and temperature to allow shutdown cooling system initiation. If feedwater cannot be restored, the operator would establish LOCA long term core cooling to continue RCS cooldown and depressurization.

In all cases, the operator is instructed to reduce RCS pressure and temperature and is precautioned not to violate the Technical Specification cooldown curves. In the case of long term restoration of feedwater, the operator could decide to close the block valves. However, the emergency procedures are clear in their instruction to continue to cool and depressurize the RCS such that the possibility of repressurization is remote.

ANS Draft Standard 58.8 (Time Response Design Criteria for Safety Related Operator Actions) prescribes 20 minutes as the minimum time before credit can be taken for operator actions from inside the control room. To preserve warm prestressing, the operator must take action by 60 minutes to continue to depressurize and cooldown the RCS. Therefore, the District concludes that the operator has sufficient instruction and time to assure that warm prestressing is preserved.

FIGURE 1

CASE 4

REACTOR VESSEL PRESSURE

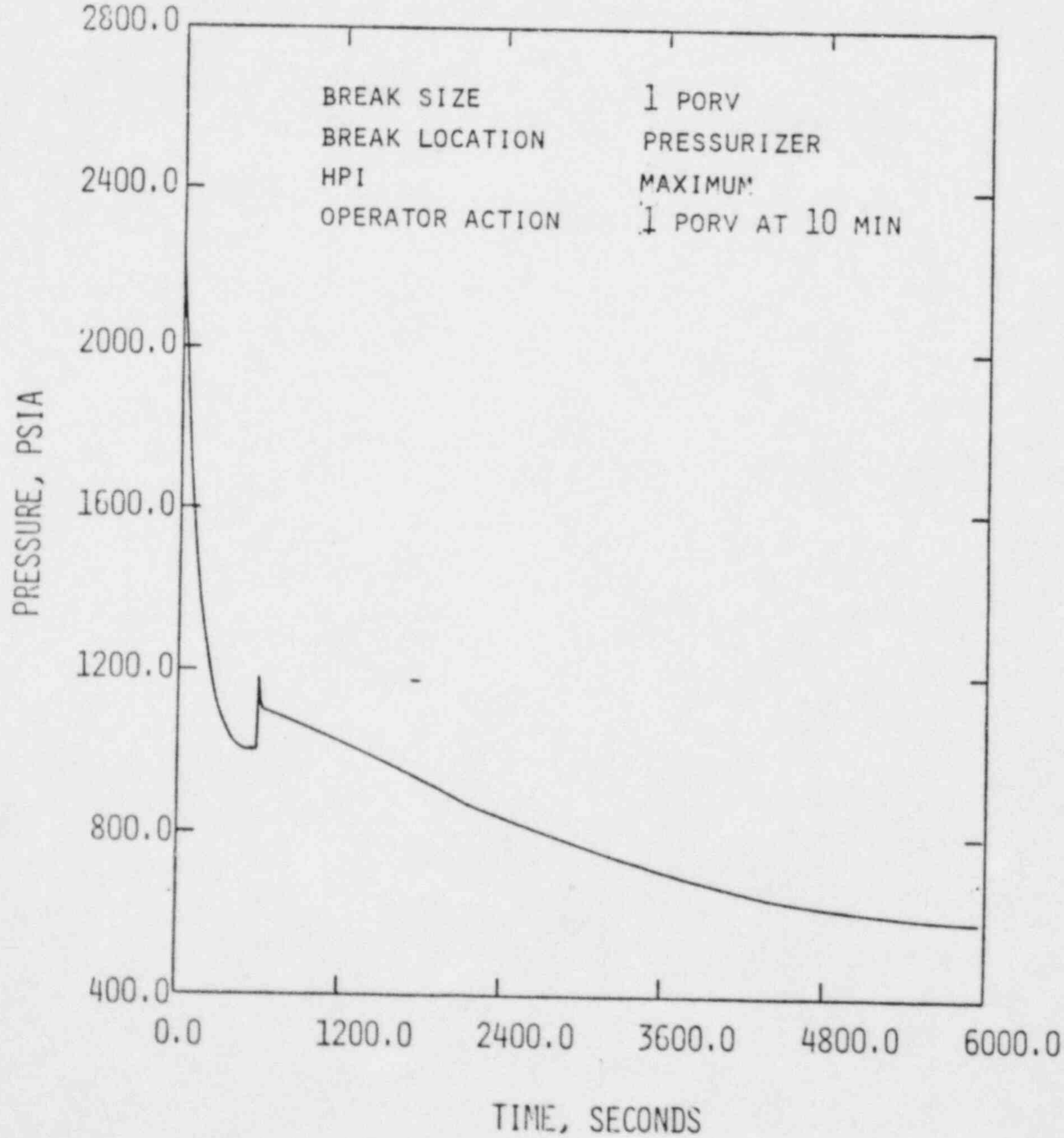


FIGURE 2

CASE 4

COLD LEG AND DOWNCOMER FLUID TEMPERATURE

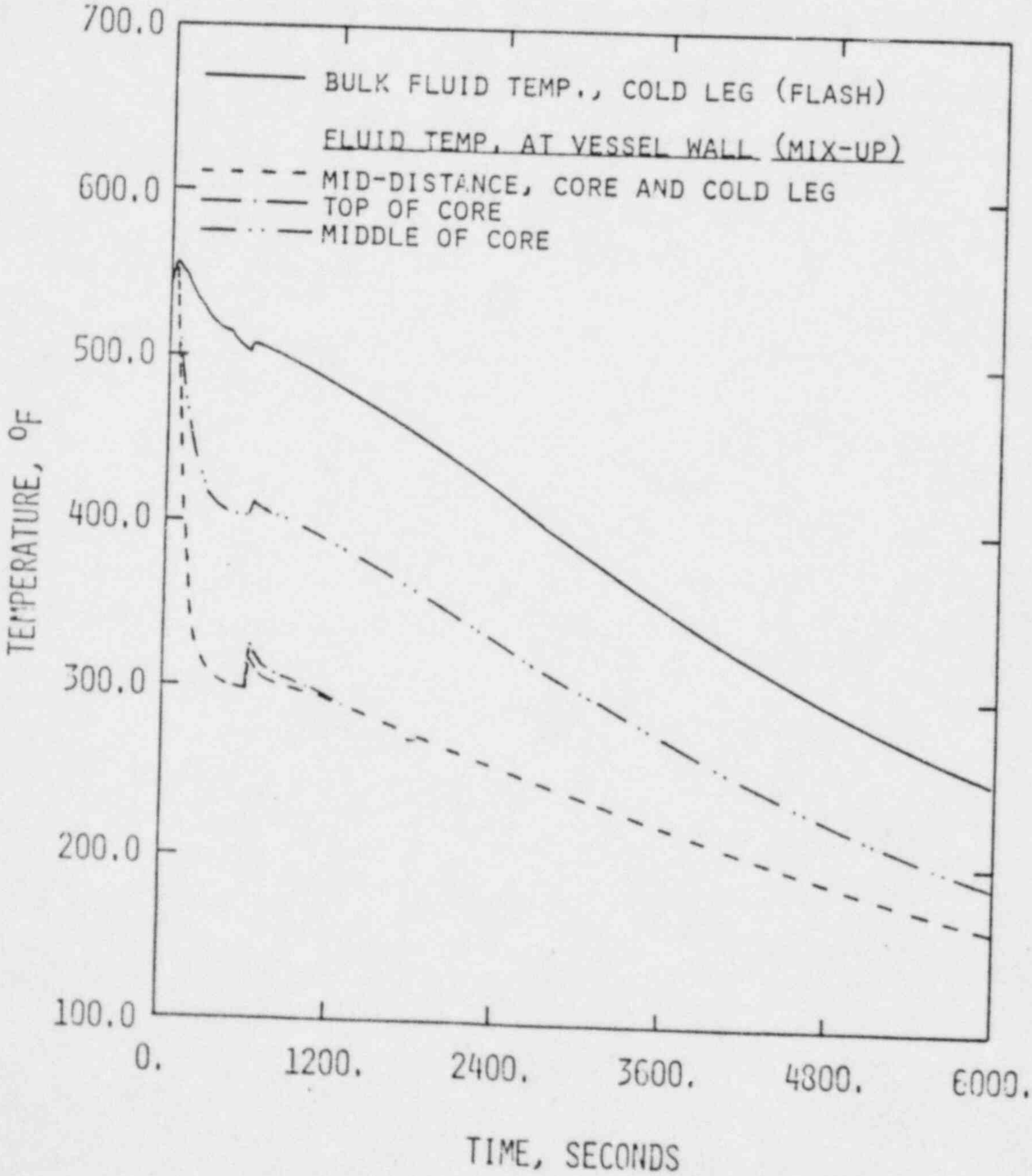
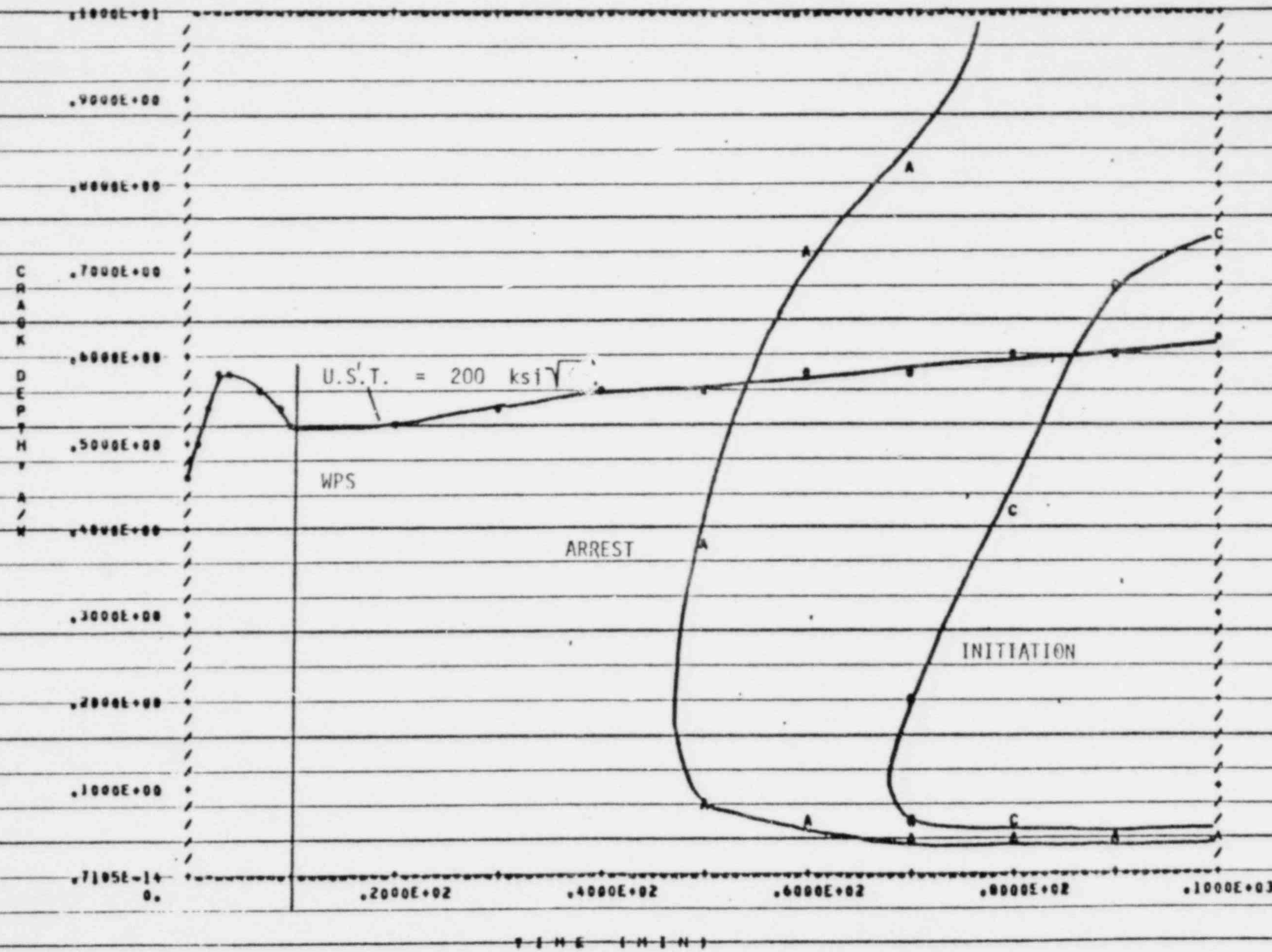


FIGURE 3

CRITICAL CRACK DEPTH VS. TIME



FLUENCE = .41E+20 N/SQ. CM
INIT RTNDT = -2 DEG. F
PCT. CU = .35
PCT. P = .012
RTNDTS = 352. DEG. F

OMAHA - CASE 4 - MIDDLE OF CORE

UPPER SHELF TOUGHNESS = 200.

4. In your evaluation, the actions described do not provide the operator with clear direction for dealing with conflicting concerns that need to be evaluated when considering the operation of HPI and the charging flow as it relates to vessel integrity and maintaining core cooling. Provide an evaluation of the need and effectiveness of procedure modifications to clearly identify the concerns in the emergency operating procedures themselves. This should be done in contrast of depending upon upgrading operator training alone.

Response

The Fort Calhoun Station emergency procedures have been modified to clearly state that the first objective is to achieve the desired degree of RCS subcooling. The emergency procedures then state that, if the desired degree of RCS subcooling and minimum pressurizer level have been achieved, the HPSI and charging pump flow are to be terminated. The procedures also state that, if the desired degree of RCS subcooling cannot be maintained, HPSI and charging pump flow are to be reinitiated. The procedures further caution the operator not to exceed the Technical Specification pressure/temperature cooldown limits. Therefore, the procedures clearly state that the first objective is to achieve adequate core cooling and, after this is achieved, the second objective is to ensure vessel integrity.

Additional evaluations were performed to evaluate the benefits of incorporating an emergency cooldown P-T limit in the operating procedures. The 200°F maximum subcooling limit curve constructed, as described at the March 3, 1982 NRC-CEOG meeting, was used as the basis for this evaluation. It was found that the conditions for minimum subcooling to assure no core voiding are achieved early in an MSLB transient, such that prompt action to keep the system below 200°F maximum subcooling would effectively prevent any repressurization. The MSLB transients reported to date show repressurization would begin about 6 to 8 minutes after start of the event. Repressurization to the HPSI pump head is observed about 10 to 12 minutes into the event. If action is taken to bring the system below the 200°F subcooling limit prior to repressurization to the HPSI pump head, then many additional EFPY of acceptable performance can be demonstrated. The difference in EFPY between the HPSI pump head pressure and the pressure achieved at 30 minutes is only 2 to 3 EFPY; however, action to bring the system below 200°F subcooling later in the transient would have the benefit of ensuring depressurization.

Analyses have shown that the current emergency procedure criteria for HPSI and charging pump termination are met at approximately the same time as the 200°F subcooling criteria. Therefore, the same beneficial effects of the 200°F subcooling criteria should be obtained using the current emergency procedure criteria.

5. The Ft. Calhoun steam line break analyses assumes that the operator trips reactor coolant pumps in 30 seconds, and reduces high pressure injection and charging flow to control plant pressure. Provide an evaluation of the sensitivity of the transient to the time assumed for operator action.

Response

A review of the pressure-temperature transients with RCP trip times of 30 seconds and 5 minutes shows that the cooldown transient is less severe for the later RCP trip case. Fracture mechanics evaluations performed using forced convection heat transfer coefficients during the pumps-on portion of the cooldown transient confirm that the early pump trip case is more governing than the later pump trip case. Therefore, the assumption of early pump trip time reported in the 150-day response was conservative and continued operation of the RCP's is beneficial with respect to PTS during any excess heat extraction event.

Concerning Probabilistic Risk Assessment (PRA) of Overcooling Transients

6. Provide existing documentation or references of such documentation related to PRAs which would provide insight into the probabilities of overcooling events at your plant.

Response

Concerning your request for probabilistic risk assessment of potential overcooling events, a review of available data was all that could be done in the short time available to respond. A comprehensive list of possible PTS scenarios was considered including different types of initiating events at different plant operating conditions. Specific event-plant condition combinations were chosen for detailed study based on their being judged to have a high likelihood to lead to the most severe PTS sequences. The sequence of events for each of the selected combinations was determined using the sequence tables and diagrams in FSAR's and CEN-128. Probabilities were determined for all logical, relevant scenarios. The scenarios which resulted from this effort were categorized as Moderate Frequency (50% probability of occurring in any one year), Infrequent (50% probability of occurring once during plant 40 year lifetime), and Limiting Fault -1, -2, or -3 (low, very low, or exceeding low probability of occurring during plant 40 year lifetime).

In summary, no Moderate or Infrequent events were determined among the scenarios considered. The MSLB initiating event was categorized as a Limiting Fault -3 event. There were a few Limiting Fault -1 and -2 events identified, but none were judged to represent a more challenging PTS event than the lower probability MSLB event.

Concerning Overcooling Transients

7. Review the operating history at your plant and identify all overcooling events as well as those events which could have become overcooling events if not mitigated by plant controls or operator actions. Provide a summary of each identified event.

Response

Concerning your request for plant experience with overcooling events, a generic CEOG task was performed to identify events which have occurred at operating plants with a CE NSSS. Events which resulted in a cooldown rate in excess of 100°F/hr and resulted in a cooldown of at least 100°F and had a duration of more than 10 minutes were reviewed. Sixteen events which were considered possible candidates were identified by a review of seven operating plants. Six of the events selected satisfied none of these criteria; three events satisfied only the 100°F/hr criterion; one event satisfied the 100°F/hr critcrion for more than 10 minutes; and only two events satisfied all three criteria. (Detailed information on the remaining four events was not available in the time available to respond.)

Of the three events which satisfied at least two criteria, the longest duration of rapid cooldown was 19 minutes and the maximum T_{ave} temperature decrease was 107°F. None of the three events exhibited repressurization at low temperature. The enclosure provides the specific results of this review.

Pressurized Thermal Shock Precursor Events of Operating Nuclear Plants with a C-E NSSS

1.0 PURPOSE

The purpose of this report is to document Pressurized Thermal Shock (PTS) precursor events that have occurred at nuclear power plants with a C-E supplied NSSS. This report is intended to provide additional scoping information on PTS for the C-E Owners Group.

2.0 SCOPE

The scope of this report is limited to the identification of PTS precursor events that have occurred at the following seven operating nuclear power plants with a C-E supplied NSSS: Palisades, Calvert Cliffs 1 and 2, Ft. Calhoun, Arkansas Nuclear One Unit 2, Millstone 2, and St. Lucie 1. This report covers C-E operating experience from January, 1971 through February, 1982.

3.0 REFERENCES

1) C-E Reliability Data System.

2) "LER Monthly Report Sorted by Facility for Power Reactors", U.S. Nuclear Regulatory Commission, Washington, D.C.

3) NUREG-0020, "Operating Units Status Report - Licensed Operating Reactors - Data for Decisions", U. S. Nuclear Regulatory Commission, Washington, D.C.

4) "Nuclear Power Experience", Nuclear Power Experiences, Inc.

5) Herbst, J. J., Paggen, V.A., "Combustion Engineering Availability Program for Nuclear Steam Supply Systems", presented at 33rd Annual Technical Conference of the American Society for Quality Control Nuclear Division, May 14 - 16, 1979, Houston, Texas.

6) MN-82-08, J. B. Randazza (Maine Yankee) letter to R. A. Clark (NRC), dated January 21, 1982.

7) A. E. Lundvall (BG&E) letter to D. G. Eisenhut (NRC), dated January 28, 1982.

8) LIC-82-029, W. C. Jones (OPPD) letter to T. M. Novak (NRC), dated January 18, 1982.

4.0 BACKGROUND

In March, 1982 the NRC requested that the C-E Owners Group provide additional material to support the CEOG position on PTS. One specific item requested by the NRC was a list of PTS precursor events that have occurred at nuclear plants with a C-E supplied NSSS. This report provides that requested information.

5.0 METHODOLOGY

Identification of actual PTS and PTS precursor events at operating nuclear power plants involves determining the conditions that existed in a plant during each transient at that plant and comparing these conditions to a predetermined set of criteria for selecting PTS precursor events.

The identification process is based on four steps including:

- o Development of PTS precursor criteria
- o Research and preliminary screening of events
- o Confirmation of event details
- o Identification of PTS precursor events

This process is further described below.

5.1 Step 1 - PTS Precursor Criteria Development

The first step in the process was to select criteria for judging whether or not a specific event was a Pressurized Thermal Shock precursor. The basic PTS precursor criteria are:

(1) $>$ 100°F/hour cooldown rate, and
(2) $>$ 100°F total cooldown, and
(3) $>$ 10 mins. duration to allow for reactor vessel response.

Application of these criteria requires a certain amount of detailed information about each transient, information that, in general, is only available at the plants. The following criteria were developed for preliminary screening of transients:

(a) the transient involved excess steam flow, or
(b) the transient involved excess feedwater flow, or
(c) the transient involved a decrease in RCS pressure followed by actual safety injection flow.

5.2 Step 2 - Preliminary Screening

Available operating experience information sources (1, 2, 3, 4, 5) were reviewed to identify events at C-E plants which met the preliminary screening criteria for PTS precursor events. A brief description was written for each such event. This preliminary review covered a total of approximately 49 plant years of operating experience.

5.3 Step 3 - Acquire Detailed Event Data

Detailed information was requested from the appropriate utility for each event selected in the preliminary screening. Confirmation of potential PTS precursor events was requested directly from the plants by C-E personnel.

5.4 Step 4 - Identify PTS Precursor Events

Using the detailed information acquired in Step 3, each potential PTS precursor event was evaluated against the basic PTS precursor criteria to see if it was a PTS precursor. A summary of events which met the basic PTS precursor criteria are given in Table 2.

6.0 RESULTS

Sixteen potential precursor PTS events were identified in the screening of operating experience data for seven operating plants with a C-E supplied NSSS. Table 1 contains a brief description of each of these events. Detailed information was provided for the potential PTS precursor events by the utilities. Evaluation of the detailed event information against the basic PTS precursor criteria produced the following results:

(a) 6 events met none of the three criteria,
(b) [illegible] events met one criterion, temperature drop > 100°F/hour,
(c) 1 event met two of the criteria, temperature drop >100°F/hour and duration > 10 minutes,
(d) 2 events met all three criteria,
(e) 4 events do not contain sufficient information to evaluate.

Table 2 provides additional data for the two events that met the PTS precursor criteria.

The conclusion of this study is that in 49 plant years of operating experience, nuclear plants with C-E supplied NSSSs have experienced only two events that met the basic PTS precursor selection criteria. Both events were significantly less severe, in terms of temperature drop, than the cases analyzed for the 150 day submittals (7, 8, 9). In neither case did the plant repressurize as a result of the event.

Table 1

POTENTIAL PTS EVENTS
FROM PRELIMINARY SCREENING

PLANT	DATE	DESCRIPTION
Palisades	12/16/78*	Reactor tripped on low steam generator (SG) water level. Main feed pump "B" failed to trip causing overfeed transient. RCS cooldown caused a safety injection (SI). Feed pump subsequently tripped by operator.
Palisades	02/01/79*	Inadvertent trip of an RCP by operator caused a reactor trip. Overfeed event, apparently caused by a failed open main feedwater regulating valve, caused an RCS cooldown. Safety Injection resulted. Feedwater regulating valve manually closed and feed pump tripped to terminate transient.
Calvert Cliffs-1	05/10/75	During full power testing two turbine bypass valves stuck open. Resultant RCS cooldown caused a SI. RCS temperature reportedly decreased approximately 100°F over several minutes. Approximately 2500 gallons injected over 10 minutes. Transient terminated when operators were dispatched and manually closed the stuck open turbine bypass valves.
Ft. Calhoun	04/74	An inadvertent loss of main feedwater caused a reactor trip on low SG level. A turbine bypass valve was subsequently opened to facilitate heat removal. Overfeeding of one steam generator occurred due to a stuck open feedwater regulating valve. RCS cooldown caused a SI. Natural circulation cooldown initiated when SIAS isolated CCW to RCP seals. Transient terminated by manually closing the affected valves.
ANO-2	12/27/78	A main steam relief valve lifted and failed to reseat during turbine roll at near hot zero power (HZP) caused a 107°F RCS cooldown over 52 minutes. Relief valve reseated after 1 hour blowdown lowered pressure.

Table 1 (Continued)

POTENTIAL PTS EVENTS
FROM PRELIMINARY SCREENING

PLANT	DATE	DESCRIPTION
ANO-2	01/79	Similar to ANO-2 12/27/78 event. Main steam relief lifted and failed to reseat. Reactor manually tripped but blowdown continued for 1 hour. Valve subsequently reseated without operator action.
Ft. Calhoun	12/77	During RCS cooldown, a SI occurred because SIAS was not blocked prior to reaching actuation setpoint. SIAS subsequently reset.
Palisades	Precommercial	During precommercial testing, two RPS breakers were opened simultaneously causing PORVs to open. Safety Injection resulted. The control room operator shut the PORV block valves to terminate the transient.
Millstone	03/80*	Loss of a main feedwater pump caused a reactor trip on low SG water level. The "B" condenser dump valve stuck open on the trip. Steam dump control was placed in manual and the dump valve was closed to terminate the transient.
Calvert Cliffs-2	12/76	During operator training at 19% power, two SG overfeeding events occurred, each apparently resulting in an RCS cooldown. Transient terminated by manual reactor trip and manual control of the feed regulating valves.
Millstone-2	02/26/76	A dropped rod occurred from 80% steady state power. Erratic control of SG water levels in manual apparently caused a reactor trip. Main feedwater ramped back to 50% of normal full flow, turbine runback apparently did not occur and the "A" steam dump valve stuck open momentarily. The feedwater regulating valves were shut in manual control and the steam dump reseated without operator intervention to terminate the transient.

Table 1 (continued)

POTENTIAL PTS EVENTS
FROM PRELIMINARY SCREENING

PLANT	DATE	DESCRIPTION
Calvert Cliffs-2	04/12/81	While returning to full power after condenser repairs, operator inadvertently overborated RCS. Reactor power sharply decreased. Reactor reportedly tripped due to instabilities in SG water level control.
Calvert Cliffs-2	09/21/81	Reactor manually tripped at full power in response to a break in a main feedwater line. Feedwater transient not seen by steam generators.
Millstone-2	01/02/81*	The reactor tripped from full power due to the loss of one 125V DC bus. Turbine trip was delayed approximately 30 seconds, resulting in an RCS cooldown. Turbine manually tripped at local control board to terminate transient.
ANO-2	01/29/80	Turbine was tripped from 100% load for power ascension tests. Reactor tripped on low SG level due to shrink effect. One steam dump valve stuck open causing a cooldown. A pressurizer spray valve also stuck open causing RCS pressure to decrease and SIAS actuation. The steam dump was manually shut and a containment entry was made to "gag" shut the spray valve.
Palisades	02/04/82	Reactor tripped on thermal margin/low pressure during a rapid power de-escalation following loss of "A" cooling tower pump. Spray valves opened, secondary safeties opened and steam dump valves opened. Secondary safeties and steam dump valves subsequently closed after extensive blowdown without operator intervention.

*Insufficient information to evaluate.

TABLE 2

EVENTS AT PLANTS WITH C-E NSSSs
MEETING PTS PRECURSOR CRITERIA

Plant	Date	Maximum Rate of Temperature Decrease (Tave)	Duration of Maximum Temperature Change Rate	Total Temperature Decrease (Tave)	RCS Pressure Decrease	Repressurize at Low Temperature	Number of Criterion Met
Ft. Calhoun	4/74	330°F/hr.	19 Minutes	107°F	700 psig	No	3
Arkansas 2	12/78	156°F/hr.	10 Minutes	107°F	~900 psig	No	3

Criterion 1:
$>100^{\circ}$F/hr.
cooldown rate

Criterion 3:
>10 minutes
event duration

Criterion 2:
$>100^{\circ}$F cooldown
of RCS