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AUTH.NAME AUTHOR AFFILIATION
VAN BRUNT,E.E. Arizona Public Service Co.
RECIP.NAME RECIPIENT AFFILIATION
KNIGHTON,G.W. Licensing Branch 3

SUBJECT: Forwards for review as proposed FSAR changes,proposed CESSAR changes re justification for LPSI & HPSU pump flow reductions,supplementing 841218 ltr.

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Arizona Public Service Company

ANPP-31535-EEVB/TFQ
December 19, 1984

Director of Nuclear Reactor Regulation
Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Post-FDA Proposed CESSAR Changes
File: 84-056-026; G.1.01.10

References: (A) Letter from E. E. Van Brunt, Jr., APS, to G. W. Knighton, NRC, dated December 18, 1984; Subject: Post-FDA Proposed CESSAR Changes.
(B) Letter from A. E. Scherer, CE, to D. G. Eisenhut, NRC, dated December 5, 1984 (LD-84-070); Subject: CESSAR Amendment 10.
(C) Letter from A. E. Scherer, CE, to D. G. Eisenhut, NRC, dated December 5, 1984 (LD-84-071); Subject: High Pressure Safety Injection Flow.

Dear Mr. Knighton:

Reference (A) requested that a number of previous proposed CESSAR changes be reviewed as proposed PVNGS FSAR changes. We wish to supplement reference (A) with those proposed CESSAR changes which we had understood to be reviewed on the CESSAR Docket (STN 50-470). These changes were transmitted by references (B) and (C), and are attached for your use.

This request is necessary, since the attached CESSAR changes pertain to justification for Low Pressure Safety Injection and High Pressure Safety Injection pump flow reductions.

Please contact Mr. W. F. Quinn of my staff if you have any questions on this matter.

Very truly yours,

E. E. Van Brunt

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/TFQ/mb
Attachments

8412260218 841219
PDR ADOCK 05000528
A PDR

B001
1/1

cc: E. A. Licitra (w/a)
A. C. Gehr (w/a)
R. P. Zimmerman (w/a)

ANPP-31535

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, A. Carter Rogers, represent that I am Nuclear Engineering Manager of Arizona Public Service Company, that the foregoing document has been signed by me for Edwin E. Van Brunt, Jr., Vice President, Nuclear, on behalf of Arizona Public Service Company with full authority so to do, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true.

A. Carter Rogers

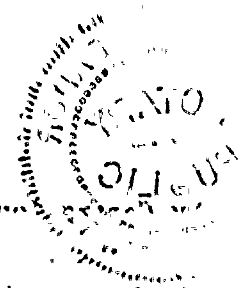
A. Carter Rogers

Sworn to before me this 19 day of December, 1984.

Notary Public

My Commission Expires:

My Commission Expires April 6, 1987



CESSAR AMENDMENT 10

EXCERPTS FROM DECEMBER 5, 1984

LETTER FROM A. E. SCHERER, CE, TO

D. G. EISENHUT, NRC (LD-84-070).

TABLE 6.3.3.3-1

SAFETY INJECTION PUMPS MINIMUM DELIVERED FLOW TO RCS

(Assuming One Emergency Generator Failed)

RCS Pressure (psig)	Flow Rate Per Injection Point*, (gpm) A1	A2	B1	B2
1775.0	0	0	0	0
1650.0	50.0	50.0	50.0	50.0
1440.0	100.0	100.0	100.0	100.0
1270.0	125.0	125.0	125.0	125.0
1095.0	150.0	150.0	150.0	150.0
865.0	175.0	175.0	175.0	175.0
605.0	200.0	200.0	200.0	200.0
310.0	225.0	225.0	225.0	225.0
200.0	234.0	234.0	234.0	234.0
130.0 ~~150.0~~	581.0 ~~1118.0~~	581.0 ~~1118.0~~	240.0 ~~238.0~~	240.0 ~~238.0~~
100.0	1282.0 ~~1743.0~~	1282.0 ~~1743.0~~	243.0	243.0
50.0	1884.0 ~~2166.0~~	1884.0 ~~2166.0~~	246.0	246.0
0	2357.0 ~~2500.0~~	2357.0 ~~2500.0~~	250.0	250.0

---

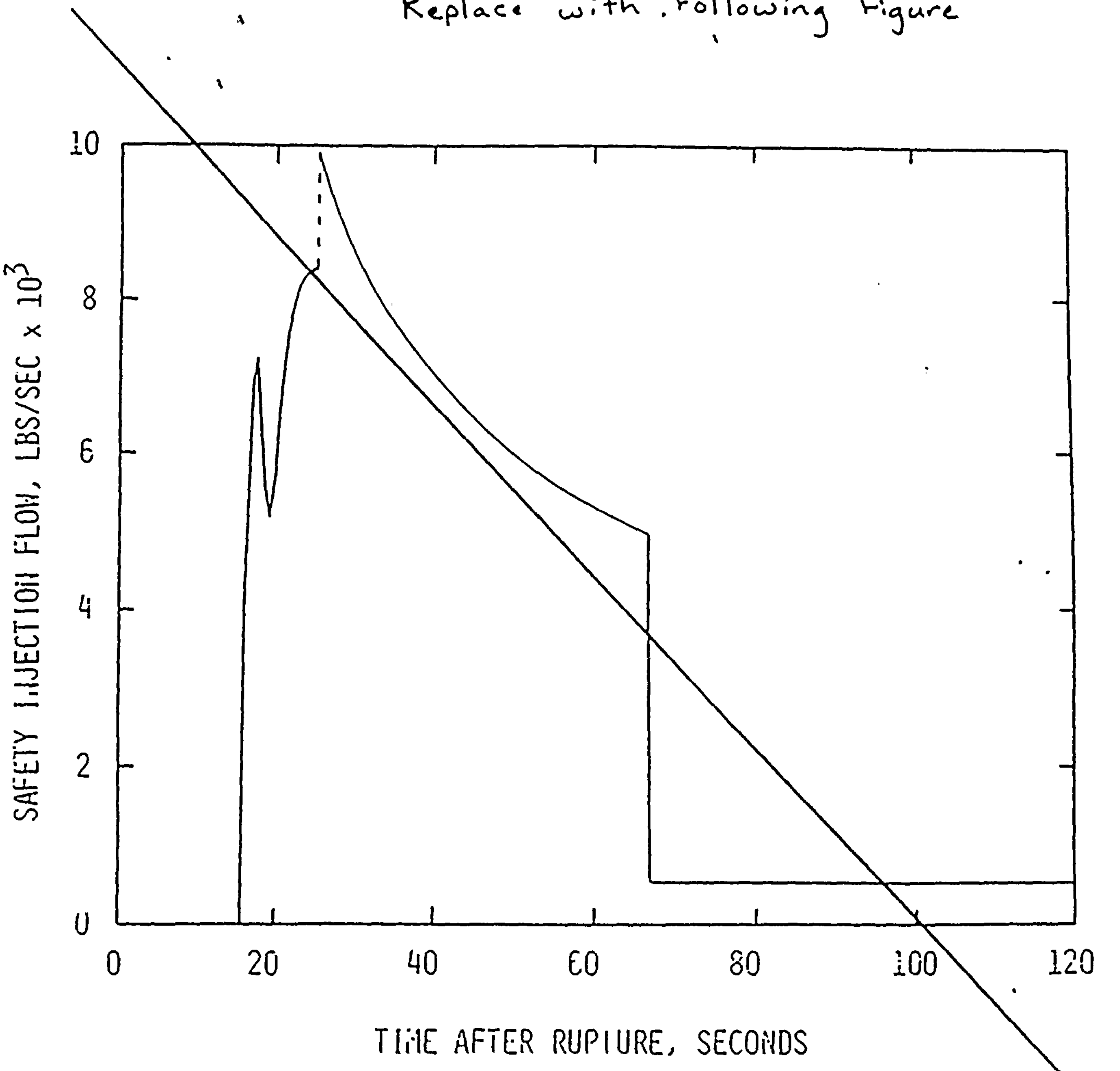
* Injection Point A1 is assumed to be attached to the broken pump discharge leg.

TABLE 6.3.3.3-2

GENERAL SYSTEM PARAMETER AND INITIAL CONDITIONS
SMALL BREAK ECCS PERFORMANCE ANALYSIS

Quantity	Value	Units
Reactor Power Level (102% of Nominal)	3876	MWt
Average Linear Heat Rate (102% of Nominal)	5.6	kw/ft
Peak Linear Heat Rate	15.0	kw/ft
Gap Conductance at Peak Linear Heat Rate	1497	btu/hr-ft²-°F
Fuel Centerline Temperature at Peak Linear Heat Rate	3681	°F
Fuel Average Temperature at Peak Linear Heat Rate	2319	°F
Hot Rod Gas Pressure	1187	psia
Moderator Temperature Coefficient at Initial Density	0.0	Δρ/°F
System Flow Rate (Total)	164.0×10^6	lbs/hr
Core Flow Rate	159.1×10^6	lbs/hr
Initial System Pressure	2250	psia
Core Inlet Temperature	565	°F
Core Outlet Temperature	623	°F
Low Pressurizer Pressure Scram Setpoint	1600	psia
Safety Injection Actuation Signal Setpoint	1600	psia
Safety Injection Tank Pressure	608	psia
High Pressure Safety Injection Pump Shutoff Head	1775	psig
Low Pressure Safety Injection Pump Shutoff Head	142. ~~[illegible]~~	psig

Replace with following figure

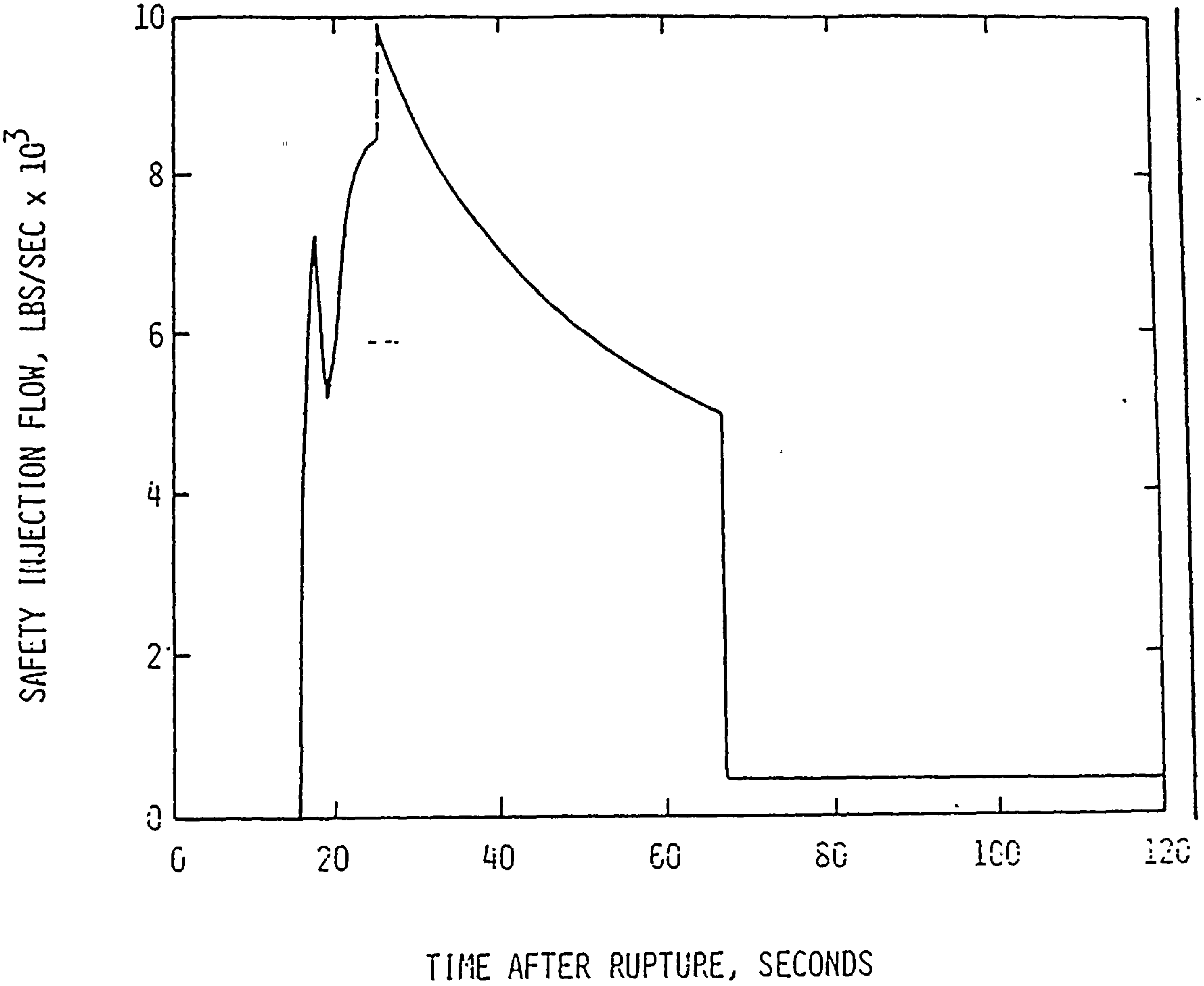


~~Amendment No. 1~~
~~July 16, 1981~~

C-E SYSTEM 80

1.0 x DOUBLE ENDED GUILLOTINE BREAK
IN PUMP DISCHARGE LEG
SAFETY INJECTION FLOW INTO INTACT DISCHARGE LEG

Figure 6.3.3.2-5L



C-E SYSTEM 80	1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG SAFETY INJECTION FLOW INTO INTACT DISCHARGE LEG	Figure 6.3.3. 2-51

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 99297

POWER SYSTEMS

STN 50-470F

December 5, 1984
LD-84-071

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: High Pressure Safety Injection Flow

Dear Mr. Eisenhut:

In an effort to provide a suitable technical specification margin for High Pressure Safety Injection (HPSI) pump performance for the first System 80™ plant, a re-analysis of the most limiting small break Loss Of Coolant Accident (LOCA) has been performed. This small break (0.05 ft^2 cold leg) was selected as the basis for determining the effect of reduced HPSI pump delivery for the following reasons.

(1) Large break LOCAs are not influenced by HPSI flow.

(2) This break size and location (0.05 ft^2 cold leg) is the most limiting small break.

(3) Reduced HPSI pump performance has no impact on the consequences of the non-LOCA Chapter 15 safety analyses.

A comparison of the previous peak clad temperature and two-phase mixture height in the core is attached (Figures 1 and 2). Also attached is a CESSAR change that is provided for your review. It will be incorporated into CESSAR in the next amendment.

A review of Figures 1 and 2 indicates tht the maximum peak clad temperature for this break size increased from 1557°F (from previous CESSAR analyses) to 1630°F. This increase is attributed to the slightly longer period of core uncovery resulting from the decrease in HPSI flow delivered. This small break analysis is still conservatively bounded by the most limiting large break LOCA peak clad temperature (2169°F occurs in a 1.0 ft^2 double-ended cold leg guillotine break).

In summary, a CESSAR change is forwarded to reflect a reduced HPSI pump flow. This change was necessary due to as-built conditions in the first System 80 plant. A re-analysis of the most limiting small break LOCA demonstrates that system performance remains well within the acceptance criteria of

10 CFR 50.46. Additionally, the higher resulting peak clad temperature remains at least 500°F below the limit case large break LOCA.

The attached change will be included in a future amendment to CESSAR. If you have any questions or comments, feel free to call me or Mr. G. A. Davis of my staff at (203) 285-5207.

Very truly yours,

COMBUSTION ENGINEERING, INC.

G A Davis for

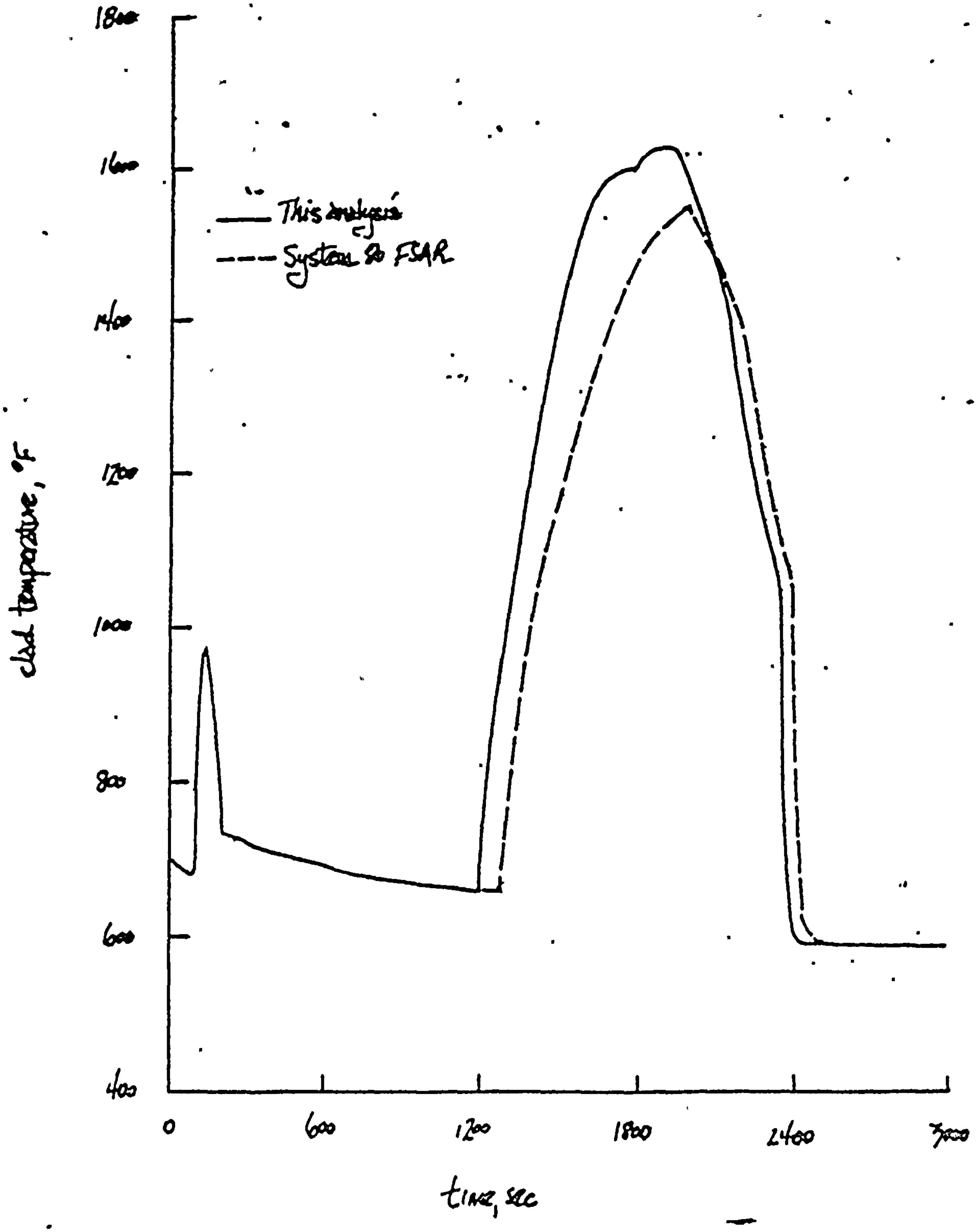
A. E. Scherer
Director
Nuclear Licensing

AES:las
Attach.
cc: P. Moriette

FIGURE 1

0.05 FT[2] BREAK -- REDUCED HPSI PUMP DELIVERY

PEAK CLAD TEMPERATURE



The four safety injection tanks (SITs) are piped so that each SIT feeds a single cold leg injection point. Thus:

a. for a break in the pump discharge leg, the SIT flow credited is 100% of the flow from three SITs. The remaining SIT is assumed to spill out the break.

b. for breaks in other locations, the SIT flow credited is 100% of four SITs.

Table 6.3.3.3-1 presents the high and low pressure safety injection pump flow rates assumed at each of the four injection points as a function of reactor coolant system pressure. INSERT(A) of next page

6.3.3.3.3 Core and System Parameters

The significant core and system parameters used in the small break calculations are presented in Table 6.3.3.3-2. The peak linear heat generation rate (PLHGR) of 15.0 kw/ft was assumed to occur 15% from the top of the active core. A conservative beginning-of-life moderator temperature coefficient of 0.0 $\Delta\rho/°F$ was used in all small break calculations.

The ECCS performance analyses as performed, do not account for steam generator tube plugging which may occur over the plant's lifetime.

The initial steady state fuel rod conditions were obtained from the FATES[7] computer program. Like the large break, the small break analyses employed a hot rod average burnup which maximized the amount of stored energy in the fuel. Since the small break analysis used a higher PLHGR than did the large break analysis (15.0 kw/ft vs 14.0 kw/ft) the fuel rod parameter values given in Table 6.3.3.3-2 differ from those on Table 6.3.3.2-2.

Because the large break results are always more limiting than the small break results, the small break analysis is run at a higher PLHGR to prevent requiring a reanalysis should the large break results improve. Since the small break results are goverened mainly by the core liquid level transient (see Results Section below) which is a function of the total core decay heat generation rate, the higher PLHGR does not significantly affect the small break results.

6.3.3.3.4 Containment Parameters

The small break analysis does not credit any rise in containment pressure. Therefore, other than the initial containment pressure, which is assumed to remain constant, no containment parameters are employed for this analysis. The initial containment pressure was assumed to be 0.0 psig.

6.3.3.3.5 Break Spectrum

Six breaks were analyzed to characterize the small break spectrum. Five breaks, ranging in size from 0.5 ft^2 to 0.02 ft^2 were postulated to occur in the pump discharge leg. The 0.5 ft^2 break was also analyzed for the large break spectrum (Section 6.3.3.2) and is defined as the transition break size[3]. One break, equal in area to a fully open pressurizer safety

INSERT(B) of next page

INSERT A

for the six break spectrum analysis identified in paragraph 6.3.3.3.5. Table 6.3.3.3-1A presents the safety injection (SI) pump flow rates used in an alternate analysis of the limiting small break LOCA, the 0.05 ft^2 break in the reactor coolant pump discharge leg. This break was reanalyzed to demonstrate the acceptability of a small reduction in the SI pump flowrate.

INSERT B

The 0.05 ft^2 break which was determined to be the limiting break size and the most sensitive to the SI pump flow capacity was also analyzed using the reduced SI pump flow discussed in paragraph 6.3.3.3.2.

valve, (.03 ft^2) was postulated to occur in the top of the pressurizer. Table 6.3.3.3-3 lists the various break sizes and locations examined for this analysis.

6.3.3.3.6 Results

The transient behavior of important NSSS parameters is shown in the figures listed in Table 6.3.3.3-4. Table 6.3.3.3-5 summarizes the important results of this analysis. Times of interest for the various breaks analyzed are presented in Table 6.3.3.3-6. A plot of peak clad temperature (PCT) versus break size is presented in Figure 6.3.3.3-7. The 0.05 ft^2 break results in the highest clad temperature ~~(1557°F)~~ of the small breaks analyzed~~, which is over 600°F lower than that reported in Section 6.3.3.1 for the limiting large break.~~ The break resulting in the next highest PCT of the small break spectrum is the 0.2 ft^2 break with a PCT of 1030°F.

It is important to note the differences in the transient behavior of these two break sizes, because each characterizes different controlling features of small breaks. The larger breaks (between 0.2 ft^2 and 0.5 ft^2) temperature transients are terminated by the action of the safety injection tanks (SIT) whereas the temperature transients for the smaller breaks (< 0.05 ft^2) are terminated solely by the high pressure safety injection pump (HPSIP) prior to the actuation of the SITs. For the intermediate break sizes (approximately 0.2 ft^2 to 0.05 ft^2) both the SITs and HPSIP play an important part in terminating the transient, with the HPSIP becoming more important as the break size decreases.

As shown in Figure 6.3.3.3-7, PCT as a function of break size remains fairly constant until the 0.2 ft^2 break. Then the PCT rises for the 0.05 ft^2 and then falls for the 0.02 ft^2 break. This rise and fall in PCT can be adequately predicted by observing the transient behavior for breaks less than or equal to 0.2 ft^2.

The peak clad temperature is predictably affected by:

1) Time of initial core uncovery,

2) Depth of core uncovery, and

3) Duration of core uncovery.

As the break size becomes progressively smaller than 0.2 ft^2, the inner vessel two phase level follows a definite pattern:

1) The time of initial core uncovery is later,

2) The depth of core uncovery is less,

3) The time of core uncovery becomes longer, and,

4) The actuation of the SITs is later during the period of core uncovery and eventually does not occur.

INSERT C

The .05 ft^2 case yeilds a peak clad temperature of 1557°F based on the SI pump flow capacities of Table 6.3.3.3-1 and 1630°F based on the SI pump flow capacities of Table 6.3.3.3-1A. In either case the result is more than 500°F higher than the other small break cases presented yet more than 500°F below the limiting large breaks reported in Section 6.3.3.1.

This trend continues until the core does not uncover at all. For System 80 this occurs for a break size between 0.05 ft^2 and 0.02 ft^2 (and for all smaller breaks).

As the break size decreases, both the later time of initial core uncovery and its shallower depth tend to mitigate the temperature transient. However, the increased duration of uncovery acts in the opposite direction. In progressing from the 0.2 ft^2 break to 0.05 ft^2 break the increased duration dominates and therefore the peak clad temperatures rise. This trend continues until a break size is reached, typified by the 0.05 ft^2 break, where the three parameters are balanced. For breaks smaller than this, the increase in time to initial core uncovery and the shallower depth dominate causing less severe temperature transients. This trend continues until the core does not uncover as typified by the 0.02 ft^2 break. Thus, by analyzing several break sizes over this range, the behavior of PCT versus break size can be adequately determined.

To demonstrate the conservatism associated with the small break ECCS performance results provided herein, the 0.05 ft^2 break was reanalyzed using a more realistic measure of the decay heat generation rate. As required by Appendix K to 10CFR50, the spectrum analysis employed a decay heat generation rate equal to 120% of the standard ANS curve. The reanalysis of the 0.05 ft^2 break used a decay heat generation rate equal to 100% of the ANS curve. This one change reduced the peak clad temperature ~~from 1557°F to 1020°F~~ by more than 500°F.

6.3.3.3.7 Instrument Tube Rupture

In addition to the ~~six~~ small breaks discussed above, the rupture of an in-core instrument tube was considered. A break, equal in size to a completely severed instrument tube (0.003 ft^2) was postulated to occur in the reactor vessel bottom head.

Following rupture, the primary system depressurizes until a reactor scram signal and safety injection actuation signal (SIAS) are generated due to low pressurizer pressure at 1600 psia. The assumed loss of offsite power causes the primary coolant pump and the feedwater pumps to coast down. After the 30 second delay required to start the emergency diesel and the high pressure safety injection pump, safety injection flow is isitiated to the reactor vessel. At this time an emergency feedwater pump is also started, providing a source of cooling to the steam generators. Due to the assumed failure of one diesel, only one high pressure safety injection pump and one emergency feedwater pump are available. (Four SITs and one low pressure safety injection pump are also available but do not inject due to the high RCS pressure.) The steam generator secondary sides also become isolated at this time.

The primary side depressurization continues accompanied by a rise in secondary side pressure until the secondary side pressure reaches the lowest set point of the steam generator safety relief valves. The primary system pressure continues to fall until it is just slightly greater than the secondary side pressure. At this point, the flow from the one operating HPSIP (66.3 lbm/sec) exceeds the leak flow (26.4 lbm/sec). Therefore the

Table 6.3.3.3-1A

SAFETY INJECTION PUMPS MINIMUM DELIVERED FLOW TO RCS
(Assuming one Emergency Generator Failed)

RCS Pressure psig	Flow Rate Per Injection Point* (gpm) A1	A2	B1	B2
1700	.5	.5	.5	.5
1581	51.25	51.25	51.25	51.25
1483	76.75	76.75	76.75	76.75
1349	102.75	102.75	102.75	102.75
1199	128.75	128.75	128.75	128.75
993	155.25	155.25	155.25	155.25
782	181.50	181.50	181.50	181.50
605	200.0	200.0	200.0	200.0
310	225.0	225.0	225.0	225.0
200	234.0	234.0	234.0	234.0
130	581.0	581.0	240.0	240.0
100	1282.0	1282.0	243.0	243.0
50	1884.0	1884.0	246.0	246.0
0	2357.0	2357.0	250.0	250.0

*Injection Point A1 is assumed to be attached to the broken pump discharge leg.

TABLE 6.3.3.3-2

GENERAL SYSTEM PARAMETER AND INITIAL CONDITIONS
SMALL BREAK ECCS PERFORMANCE ANALYSIS

Quantity	Value	Units
Reactor Power Level (102% of Nominal)	3876	MWt
Average Linear Heat Rate (102% of Nominal)	5.6	kw/ft
Peak Linear Heat Rate	15.0	kw/ft
Gap Conductance at Peak Linear Heat Rate	1497	btu/hr-ft^2-°F
Fuel Centerline Temperature at Peak Linear Heat Rate	3681	°F
Fuel Average Temperature at Peak Linear Heat Rate	2319	°F
Hot Rod Gas Pressure	1187	psia
Moderator Temperature Coefficient at Initial Density	0.0	Δρ/°F
System Flow Rate (Total)	164.0×10^6	lbs/hr
Core Flow Rate	159.1×10^6	lbs/hr
Initial System Pressure	2250	psia
Core Inlet Temperature	565	°F
Core Outlet Temperature	623	°F
Low Pressurizer Pressure Scram Setpoint	1600	psia
Safety Injection Actuation Signal Setpoint	1600	psia
Safety Injection Tank Pressure	608	psia
High Pressure Safety Injection Pump Shutoff Head	1775. /1700 (a)	psig
Low Pressure Safety Injection Pump Shutoff Head	142 (a)	psig

(a) Value used for the reanalysis of the 0.05 ft^2 break with the SI pump flow rates listed in Table 6.3.3.3-1A

TABLE 6.3.3.3-5

FUEL ROD PERFORMANCE SUMMARY
SMALL BREAK SPECTRUM

Break Size (ft^2)	Maximum Clad Surface Temperature (°F) [(a)]	Peak Local Zirconium Oxid. (%) [(b)]	Hot Rod Zirconium Oxid. (%) [(c)]
0.50 ft^2/PD [(d)]	954	<.0020	<.0003
0.35 ft^2/PD [(d)]	932	<.0015	<.0002
0.20 ft^2/PD [(d)]	1030	<.0041	<.0007
0.05 ft^2/PD [(d)]	1557	<.8825	<.1430
0.02 ft^2/PD [(d)]	995	<.0011	<.0003
0.03 ft^2/HL [(d)]	1012	<.0011	<.00004
0·05 ft^2/PD [(e)]	1630	<1·4126	<·2083

(a) Acceptance Criteria is 2200°F.

(b) Acceptance Criteria is 17%.

(c) Acceptance Criteria is 1.0%. Hot rod oxidation values are given as a conservative indication of core-wide oxidation.

(d) Break analyzed using the SI pump flow rates listed in Table 6.3.3.3-1

(e) Break analyzed using the SI pump flowrates listed in Table 6.3.3.3-1A.

TABLE 6.3.3.3-6

TIMES OF INTEREST FOR SMALL BREAKS
(Seconds)

Break Size (ft^2)	HPSI Pump On	LPSI Pump On	SI Tanks On	Hot Spot Peak Clad Temp. Occurs
0.50 ft^2/PD [(c)]	46.5	158.0	142.0	160.0
0.35 ft^2/PD [(c)]	50.0	244	204.0	235.0
0.20 ft^2/PD [(c)]	62.0	445	400.0	442.0
0.05 ft^2/PD [(c)]	208.0	a.	b.	2010.0
0.02 ft^2/PD [(c)]	492.0	a.	b.	437.0
0.03 ft^2/HL [(c)]	585.0	a.	b.	540.0
0.05 ft^2/PD [(d)]	212.0	a	b	1900

a. Calculation terminated before time of LPSI pump activation.

b. Calculation terminated before initiation of SI tank discharge

c. Break analyzed using the SI pump flow rates listed in Table 6.3.3.3-1

d. Break analyzed using the SI pump flow rates listed in Table 6.3.3.3-1A.