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T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444

Subject: Additional Information for the Locked Rotor/Shaft Break Event: (SRP 15.3.3; RAI 440.89 and 440.127; Reactor Systems Branch)

Dear Sir:

The enclosed Additional Information for the Locked Rotor/Shaft Break Event supplements our previous responses which were submitted in letters dated March 12, 1982, September 21, 1982 and December 7, 1982.

This open item was discussed with representatives of the Reactor Systems Branch and Accident Evaluation Branch in meetings conducted on January 10-12, 1983.

The enclosed response will be included in OL Application Amendment 49.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. DeVincentis
Project Manager

B001

ALL/fsf

cc: Atomic Safety and Licensing Board Service List

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Additional Information for the Locked Rotor/Shaft Break Event

In the event of a locked rotor/shaft break of a Reactor Coolant Pump (RCP), the remaining three RCPs will continue to run. Analysis of the breaker coordination (see response to RAI 430.55) shows the following. Under all postulated operating conditions, including maximum load of one of the 13.8 kV buses (2 RCPs, 2 Circulating Water Pumps and the 13.8 kV Substations) and minimum bus voltage, failure of one RCP (with incipient locked rotor amps) will not result in tripping of the incoming breaker to the 13.8 kV bus. Because of the separate power supply to the other 13.8 kV bus (see FSAR Figure 8.3-1), this event will have no effect on the power supply of this bus.

Off-site power will not be lost as a consequence of the event. FSAR Section 8.2.2.3 provides the results of stability studies showing that the loss of one or both of the Seabrook Units would not cause a loss of off-site power. FSAR Figure 8.3-1 is a one-line diagram of the Electrical Distribution System showing the generator circuit breaker used for isolating the generator without affecting the normal supply to the 13.8 kV bus.

During this postulated event, the maximum number of fuel rods with a minimum transient DNBR less than 1.3 (rods in DNB) occurs near the time of maximum clad temperature. Furthermore, the number of "rods in DNB" is related to the calculated maximum clad temperature. From the results shown in FSAR Sections 15.3.3 and 15.3.4, it is seen that the maximum clad temperature occurs about 2.8 seconds after turbine trip. The FSAR analysis assumes no loss of off-site power. Therefore, the remaining three Reactor Coolant Pumps (RCPs) are assumed to continue operating. Now, based on the grid stability studies described above, it has been demonstrated that off-site power (and the remaining RCPs) would <u>not</u> be lost and would be available well beyond the time of maximum clad temperature. So, Loss of AC at any time after the maximum clad temperature is predicted (about 2.8 seconds after turbine trip) would yield the same number of "rods in DNB" as the existing FSAR analysis.

If the assumption is made that all the rods with a minimum transient DNBR of less than 1.3 become failed rods, then the fraction of failed fuel is predicted to be 8% for this bounding event. However, it is the applicant's position that "rods in DNB" is an overly conservative criteria for determining fuel failure. When the evaluation procedures of NUREG-0562 are followed, <u>no</u> fuel failure is predicted. The attached material is submitted in support of this position, although no credit was taken for it.

With a conservatively assumed 8% fuel clad failure fraction, the off-site doses, as presented in FSAR Section 15.3.3.3, are well within the design limits (i.e., 10% of 10CFR Part 100 guideline values).

In the event of a failed open atmospheric steam dump valve, the off-site doses are well within the design limits specified in 10 CFR Part 100. The off-site thyroid doses, in the event of a failed open valve concurrent with the locked rotor event plus the assumed 8% fuel clad failure and the assumed loss of off-site power, are a maximum of 100 times the values presented in

FSAR Section 15.3.3.3 or 230 rems and 240 rems at the EAB and LPZ, respectively.[1] With the failure of a steam dump valve in the open position, plant procedure would call for isolation of the affected steam generator's feedwater flow with subsequent drying out of the steam generator and loss of any iodine partitioning afforded by the water volume contained above the point of primary to secondary leakage. Although some finite period of time is required for the considerable amount of water contained within a typical Westinghouse steam generator to boil off, it has been conservatively assumed that this time is small and no partitioning of iodine occurs within the steam generator.

The failure of an atmospheric steam dump valve, discussed above, is considered to be the limiting single failure associated with the locked rotor event. In addition to the sequence of events mentioned above, it is also conservatively assumed that the entire primary to secondary allowable leakage of 1 gpm occurs in the steam generator that is venting through the stuck open valve.

Corrections to the FSAR are attached and will be placed in the FSAR with the next amendment.

1 The off-site whole body gamma doses for this event are calculated to be 1.4 rem and 1.9 rem at the EAB and LPZ, respectively.

APPLICATION OF NUREG-0562 TO THE EVALUATION OF FUEL FAILURE AS A CONSEQUENCE OF A LOCKED ROTOR ACCIDENT IN A W PWR

INTRODUCTION

Analysis of locked rotor (LR) accident under the assumption that rods in DNB fail produces overly-conservative dose rates. One way to obtain more realistic results is to demonstrate that no fuel failures will occur during a LR accident, even though some of the rods may go into DNB. The material in the following sections can be used to support this conclusion. Please note that in no way does this conclusion replace the need to verify by means of peak clad temperature analysis that the fuel maintains a coolable geometry for the LR accident.

LOCKED ROTOR AND FUEL FAILURE MECHANISMS

The LR is a Condition IV occurrence. The design requirements for such faults do not state that the plant must be capable of returning to operation with the existing core. Thus we will only address the effect of DNB during a LR on the ability of the clad not to fail during the accident, during shutdown and during possible removal and inspection of the fuel after the accident. We will not address the effect of DNB during a LR on subsequent operation of the fuel rods.

Several mechanisms for cladding failure due to DNB have been identified. These are: (a) clad melting, (b) fuel pellet melting, (c) clad ballooning/bursting, (d) clad collapse, (e) clad recrystallization, and (f) excessive clad oxide. A discussion of whether any of these is a failure mechanism due to DNB caused by a locked rotor accident is discussed below.

(a) Clad Melting

The melting point[1] of Zircaloy-2 and -4 is 3360°F. This is well above any clad temperatures conservatively estimated to occur during LR for a typical plant (Figure 3). Thus, LR does not result in conditions in which clad melting is a clad failure mechanism.

(b) Fuel Pellet Melting

The melting point[1] of UO_2 during a DNB incident would cause the fuel pellet to expand. This could strain the clad to failure. Another possible failure mechanism would be contact of the molten fuel with the clad. The melting point of unirradiated UO_2 is 5080°F with a melting point reduction during irradiation of 58°F per 10,000 MWD/MTU burnup. Calculations of pellet centerline temperatures for LR conditions give maximum temperatures lower than the melting point temperatures. Thus fuel melting is not a clad failure mechanism.

(c) Clad Ballooning/Bursting

For the LR transient, the RCS pressure should increase or remain unchanged during the event. Consequently, high powered rods in DNB should have internal pressures sufficiently low compared to the system pressure that the rods will not balloon for clad temperatures predicted to occur during a LR accident. As a result, no fuel failure will occur by this mechanism during the LR event.

(d) Clad Collapse

Since most of the rods with low burnup and high $F_{\Delta H}^{N}$ will have internal pressures less than system pressure, if DNB were to occur on these rods and decrease the yield strength, the clad would collapse onto the fuel. Clad collapse onto the fuel pellets is not considered a fuel failure mechanism as long as there is no subsequent operation.

(e) Clad Recrystallization

When the zirc cladding is raised above the temperature range of 950° to 1000°F, the hardening and strengthening due to cold working is removed. This results in lower yield stresses and lower ductility for irradiated material.

Failures due to a change in the Zircaloy clad properties due to the temperature increases in the fuel rods during the LR accident should not occur during nor after the accident. Such changes would only need to be considered in subsequent operations of the core.

(f) Excess Clad Oxide

Failure due to excessive clad oxidation is considered to be a major cause of fuel rod failure during DNB[2]. This mechanism is addressed in Reference 2, and it can be shown that the clad temperatures reached during the time interval of a LR accident are insufficient to cause fuel failure. An updated discussion of this topic is presented in this section.

Van Houten[2] of the NRC reviewed the available literature on the operation of nuclear fuel rods under film boiling or dryout conditions. His review was mainly concerned with data on oxidation-related failures. The discussion below is based on his review.

Figure 1 is based on the results of out-of-pile Zircaloy-steam oxidation tests of Kassner and Chung et.al. of ANL. In these tests the cladding was subjected to a rapid heatup, an isothermal soak and a rapid cooldown. The ability of the oxidized cladding to withstand the thermal shock of the quench and/or shock of a subsequent sharp impact was then determined. The data were for a .6 mm thick cladding (W 17x17 cladding thickness is .0225" = .57 mm). The limit line for surviving a .3 Joule impact at 300°K (80°F) is given. Studies showed that if the oxidized cladding can withstand this, then the cladding is still tough enough to withstand both operating stresses when at working temperatures and the mechanical stresses of handling when at room temperatures. The line for surviving thermal shock corresponds to the quenching of the hot oxidized cladding. This line is roughly equivalent to a limit line for surviving at .03 Joule impact at 300°K (80°F). The 17% oxidation and 1477°K (2200°F) lines correspond to NRC imposed LOCA limits.

Figure 2 gives the results of inpile test data of fuel rods operated under DNB or dryout conditions. The two limit lines from Figure 1 describing the ability of the oxidized clads to withstand thermal shock or impact are also plotted. Comparisons of these lines to the data indicates that the limit lines based on out-of-pile data are effectively the same as those based on inpile data.

Figure 3 gives the W fuel clad temperature as a function of time for a typical worst type LR accident. The temperatures were calculated under conservative assumptions and assumed that the rod was in film boiling. Thus, if it were conservatively assumed that the clad temperature remains at 2000°F (1366°K, $.00073°K^{-1}$) for 10 seconds, it is seen that this gives a point on Figures 1 and 2 well below the limit lines. Thus the limiting rod in DNB during a LR accident will have a large margin to withstand both the temperature transient (heatup and quench) that occurs during the accident and the handling of the rod after the accident.

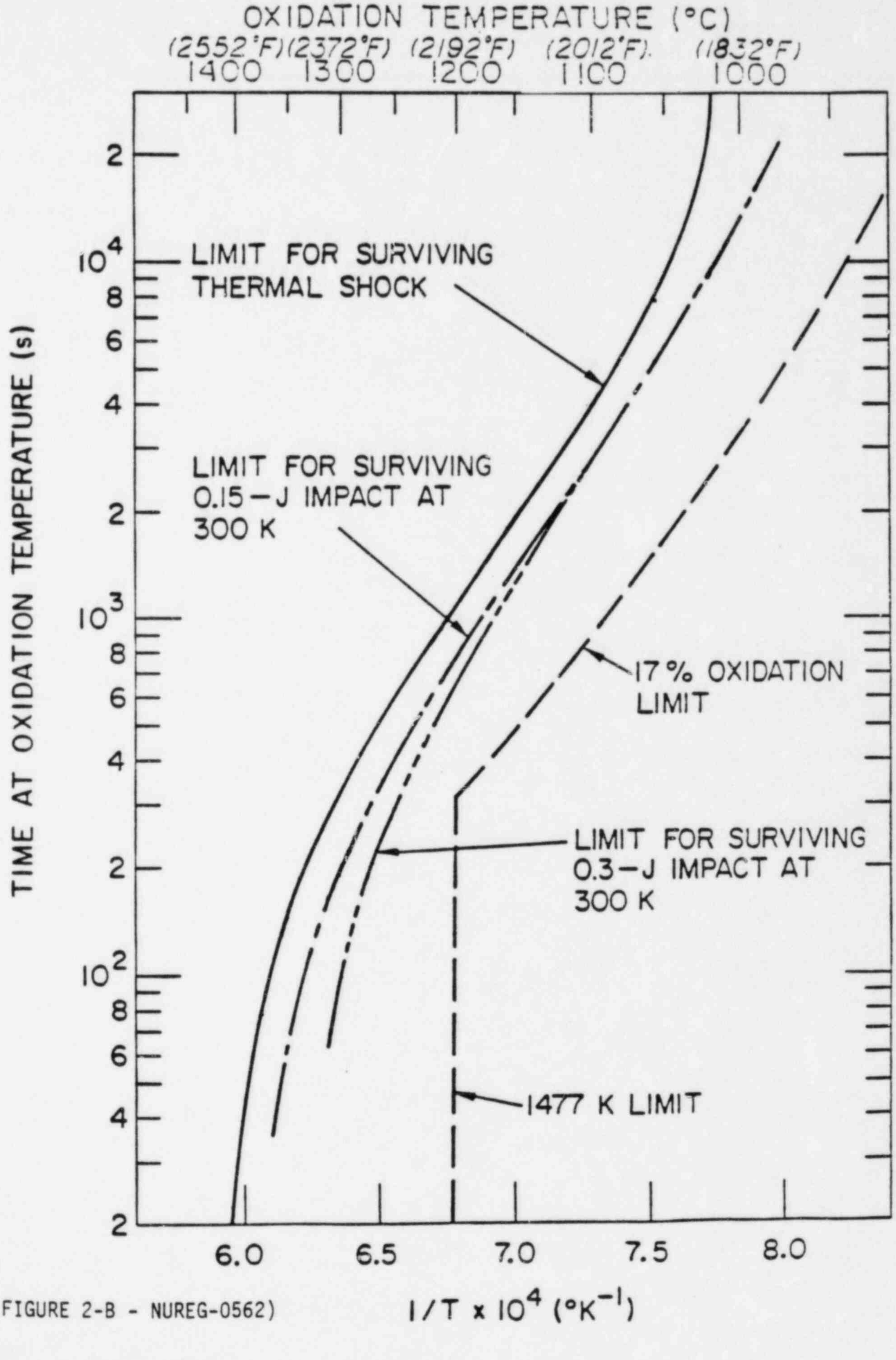
CONCLUSION

W maintains that as long as the locked rotor transient results in a time-at-temperature below the oxidized clad failure limit lines in Figures 1 and 2, no fuel failure will result. This does not replace the need to verify by means of peak clad temperature analysis that the fuel maintains a coolable geometry for the locked rotor accident.

REFERENCES

1. Engineering Department "Materials Property Manual," WCAP-8222.
2. Robert Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout," NUREG-0562, June 1979.

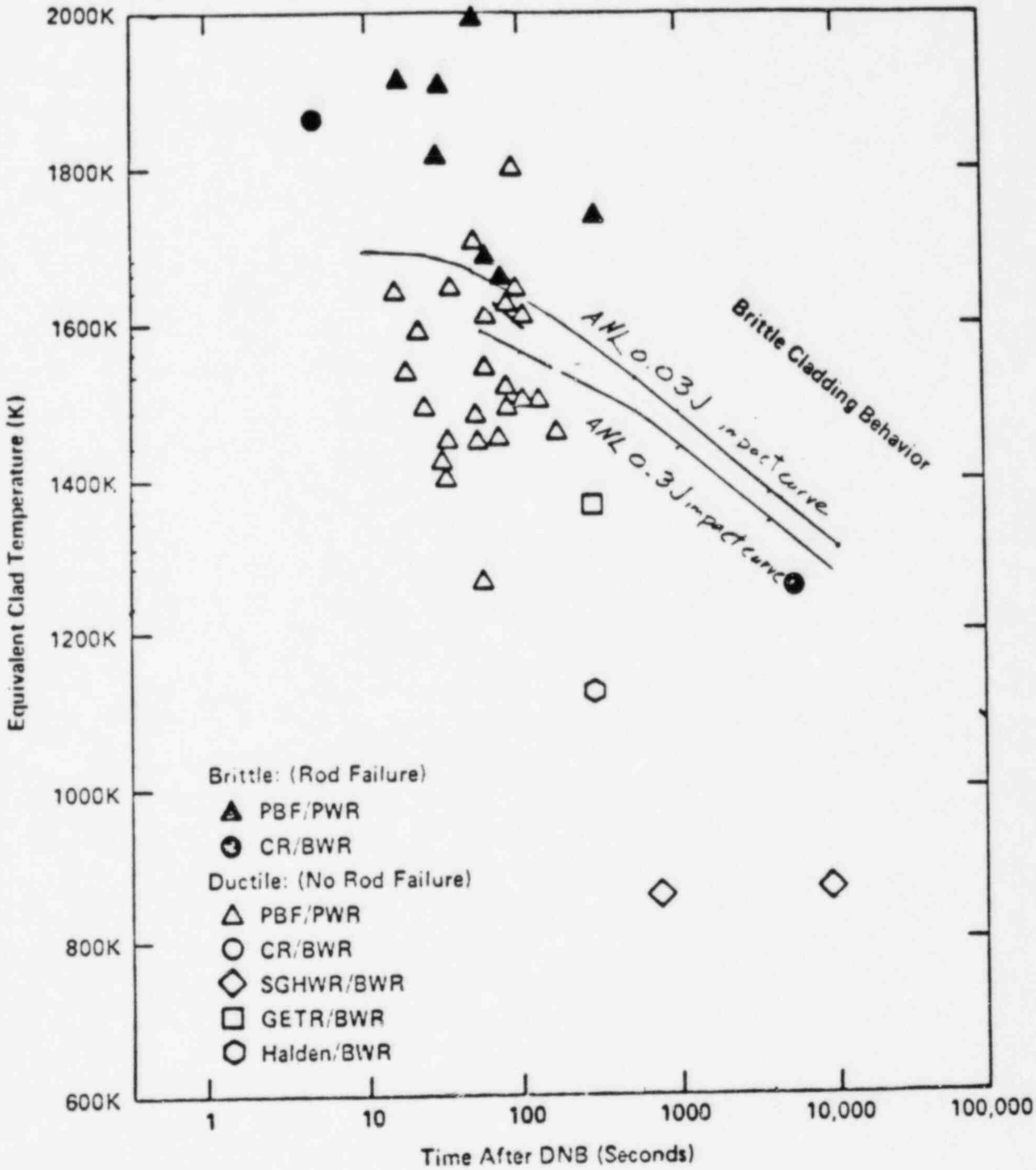
FIGURE 1



(FIGURE 2-B - NUREG-0562)

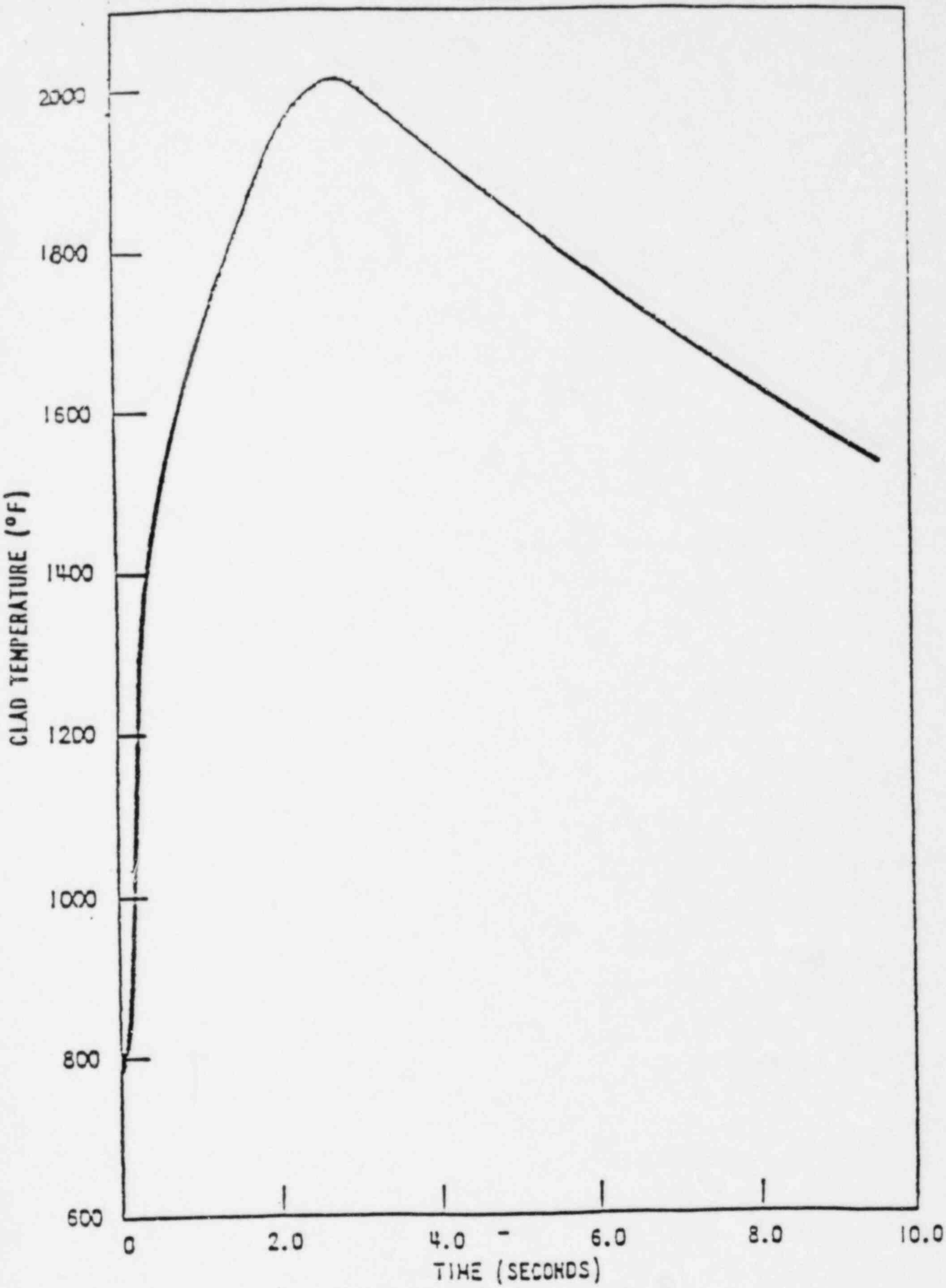
FIGURE 2

Comparison of In-Reactor Post-DNB Survival/Failure Data for Zircaloy Cladding with FBRB/ANL Zircaloy Ductile-Brittle Boundary Curve



(FIGURE 4 - NUREG-0562)

FIGURE 3



CLAD TEMPERATURE AS A FUNCTION OF TIME
FOR A TYPICAL WORST CASE LOCKED ROTOR EVENT
(All Loops Operating, One Locked Rotor)

Revised FSAR Pages For Locked Rotor Event

(e) Primary to secondary coolant system leakage rate is 1 gpm (at an average primary coolant liquid density of .71 gm/cc) for the 8-hour period during which steam releases to the atmosphere may be required. For this 8-hour period, 100% of the noble gases and 1% of the iodines in the secondary coolant system, from the primary coolant leakage, are assumed to be released with the steam to the atmosphere. Table 15.3-5 lists the activity released to the environment via the steam generator safety/relief valves.

2. Realistic Analysis

(a) Initial primary coolant activity prior to the accident corresponds to the equilibrium concentration at 0.12% clad defects (see Table 11.1-1).

(b) As a result of the accident, 0.02% of the iodines and noble gases of the core activity are released into the primary coolant system.

This release from the damaged fuel rods, plus the primary coolant activity prior to the accident, represents the source term for the realistic analysis of this accident and is listed in Table 15.3-6.

(c) For the realistic analysis, offsite power is assumed to be available. The condenser and the secondary coolant system are used to cool down the plant. The time period of plant cooldown is assumed to be 8 hours.

(d) The realistic source term in Table 15.3-6 is uniformly mixed with the mass of primary coolant. Table 15.3-7 lists the activity concentration of primary coolant after the accident.

(e) Primary to secondary coolant system leakage rate is 0.009 gpm for the 8-hour period during which the release from the secondary side is via the condenser air evacuation pump. Decontamination factors of 0.01 and 1.0 for iodine and noble gases, respectively, are assumed for the steam generator. Decontamination factors of 0.001 and 1.0 for iodines and noble gases, respectively, are assumed for the main condenser. During this 8-hour period of time, 100% of the noble gases and 0.15% of the iodines in the secondary coolant system, from the primary coolant leakage, are assumed to be released with the steam to the atmosphere. Table 15.3-7 lists the activity released to the environment via the condenser air evacuation pump.

TABLE 15.3-5

RELEASE FROM STEAM GENERATOR STEAM DUMP LOCKED ROTOR - CONSERVATIVE

Radionuclide	Primary Coolant Concentration After Accident (μCi/gm)	0-2 Hours Release (Ci)	0-8 Hours Release (Ci)
I-131	3.5 + 03*	1.1 ~~1.6~~ + 01	4.5 ~~6.4~~ + 01
132	4.8 + 03	1.6 ~~2.2~~ + 01	6.2 ~~8.7~~ + 01
133	7.4 + 03	2.4 ~~3.4~~ + 01	9.2 ~~1.3~~ + 0~~2~~1
134	7.8 + 03	2.5 ~~3.5~~ + 01	9.9 ~~1.4~~ + 0~~2~~1
135	7.0 + 03	2.3 ~~3.2~~ + 01	9.2 ~~1.3~~ + 0~~2~~1
Kr-83m	6.3 + 01	1.4 ~~2.0~~ + 01	5.5 ~~7.8~~ + 01
85m	9.6 + 02	3.1 ~~4.4~~ + 02	1.2 ~~1.7~~ + 03
85	6.5 + 01	2.1 ~~3.0~~ + 01	8.5 ~~1.2~~ + 0~~2~~1
87	1.7 + 03	5.5 ~~7.7~~ + 02	2.2 ~~3.1~~ + 03
88	2.5 + 03	7.8 ~~1.1~~ + 0~~3~~2	3.2 ~~4.5~~ + 03
89	3.1 + 03	9.9 ~~1.4~~ + 0~~3~~2	4.0 ~~5.6~~ + 03
Xe-131m	2.6 + 01	8.5 ~~1.2~~ + 0~~1~~0	3.3 ~~4.7~~ + 01
133m	1.0 + 03	3.2 ~~4.5~~ + 02	1.3 ~~1.8~~ + 03
133	7.0 + 03	2.3 ~~3.2~~ + 03	9.2 ~~1.3~~ + ~~04~~03
135m	1.4 + 03	4.5 ~~6.4~~ + 02	1.8 ~~2.5~~ + 03
135	1.5 + 03	4.8 ~~6.8~~ + 02	1.9 ~~2.7~~ + 03
138	5.7 + 03	1.8 ~~2.6~~ + 03	7.1 1.0 + 0~~4~~3

* 3.5 + 03 = 3.5 x 10^3.

TABLE 15.3-7

ACTIVITY RELEASED VIA AIR EVACUATION PUMP LOCKED ROTOR - REALISTIC

Radionuclide	Primary Coolant Activity Concentration (μCi/gm)	0-2 Hours Release (Ci)	0-8 Hours Release (Ci)
I-131	8.8 + 01*	2 ~~3~~.6 - 06	1.0 ~~1.4~~ - 05
132	1.2 + 02	3.5 ~~5.0~~ - 06	1.4 ~~2.0~~ - 05
133	1.8 + 02	5.4 ~~7.5~~ - 06	2.1 ~~3.0~~ - 05
134	2.0 + 02	5.8 ~~8.2~~ - 06	2.3 ~~3.3~~ - 05
135	1.8 + 02	5.1 ~~7.1~~ - 06	2.0 ~~2.9~~ - 05
Kr-83m	1.1 + 00	3.1 ~~4.4~~ - 03	1.2 ~~1.8~~ - 02
85m	2.5 + 01	7.6 ~~1.1~~ - 0~~1~~2	3.0 ~~4.3~~ - 01
85	5.4 - 01	1.6 ~~2.2~~ - 03	6.3 ~~8.8~~ - 03
87	4.4 + 01	1.3 ~~1.8~~ - 01	5.1 ~~7.2~~ - 01
88	6.3 + 01	1.8 ~~2.6~~ - 01	7.3 ~~1.0 + 00~~ -01
89	7.8 + 01	2.3 ~~3.2~~ - 01	9.0 ~~1.3 + 00~~ -01
Xe-131m	6.3 - 01	1.8 ~~2.6~~ - 03	7.4 ~~1.0~~ - 0~~2~~ 3
133m	2.6 + 01	7.6 ~~1.1~~ - 0~~1~~2	3.0 ~~4.3~~ - 01
133	1.8 + 02	5.1 ~~7.2~~ - 01	2.0 ~~2.9~~ + 00
135m	3.6 + 01	1.0 ~~1.5~~ - 01	4.2 ~~5.8~~ - 01
135	3.8 + 01	1.1 ~~1.5~~ - 01	4.4 ~~6.2~~ - 01
138	1.4 + 02	4.1 ~~5.7~~ - 01	1.6 ~~2.3~~ + 00

* 8.8 + 01 = 8.8 x 10^1.

TABLE 15.3-8

OFFSITE DOSES AT GIVEN SITE DUE TO LOCKED ROTOR ACCIDENT

Site*	Thyroid (Rem)	Whole Body (Rem)	Skin (Rem)
Conservative Analysis			
EAB (0-2 Hours)	2.3E+00* ~~3.3E+00*~~	4.4E-01 ~~6.1E-01~~	7.4E-01 ~~1.0E+00~~
LPZ (0-8 Hours)	~~3.3E+00~~ 2.4E+00	~~6.3E-01~~ 2.1E-01	~~1.1E+00~~ 3.6E-01
Realistic Analysis			
EAB (0-2 Hours)	1.0E-07	1.9E-05	3.3E-05
LPZ (0-8 Hours)	1.1E-07	2.1E-05	3.6E-05

* EAB: Exclusion Area Boundary; LPZ: Low Population Zone

** ~~3.3E+00 = 3.3 x 10^0 = 3.3~~
2.3E+00 = 2.3 × 10^0 = 2.3