Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

September 26, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Braidwood Generating Station Units 1 and 2
Reactor Coolant Pump Transients
NRC Docket Nos. 50-454/455 and 50-456/457

References (a): June 7, 1982 letter from T. R. Tramm to H. R. Denton.

(b): May 2, 1984 letter from T. R. Tramm to H. R. Denton.

Dear Mr. Denton:

This letter provides additional information regarding postulated reactor coolant pump locked rotor and shaft break transients for the Byron/Braidwood units. NRC review of this information should close Confirmatory Issue 30 of the Byron SER.

In reference (b), Commonwealth Edison provided advance copies of revised FSAR pages which document a recent reanalysis of the locked rotor transient. After review of these pages by the NRC, it has become apparent that other FSAR changes were also necessary.

Enclosed are advance copies of revised FSAR section 15.3.3 and Tables 15.0-9 through 15.0-12. These changes will be incorporated into the FSAR at the earliest opportunity. These changes make it clear that, although no fuel rods experience DNB, the analysis conservatively assumed DNB at the core hot spot in determining the upper limits of clad temperature and zirconium-water reaction.

Please address further questions regarding this matter to this office.

One signed original and fifteen copies of this letter and the attachments are provided for NRC review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Admininstrator

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Attachment

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TABLE 15.0-9

SECONDARY COOLANT EQUILIBRIUM IODINE ACTIVITY
BASED ON 0.1 μCi/gm OF
DOSE EQUIVALENT I-131

ISOTOPE	CONCENTRATION (μCi/gm)
I-131	0.066
I-132	0.239
I-133	0.106
I-134	0.016
I-135	0.058

TABLE 15.0-10

REACTOR COOLANT IODINE ACTIVITY BASED ON 60 μCi/gm OF DOSE EQUIVALENT I-131 AND REACTOR COOLANT NOBLE GAS INVENTORY BASED ON 1% FUEL DEFECTS

ISOTOPE	ACTIVITY (μCi/gm)
I-131	39.76
I-132	14.31
I-133	63.62
I-134	9.54
I-135	34.99
Xe-133	398.03
Xe-133m	4.39
Xe-135	8.92
Xe-135m	0.283
Xe-138	0.992
Kr-85	12.46
Kr-85m	2.97
Kr-87	1.70
Kr-88	5.24

TABLE 15.0-11

POTENTIAL OFFSITE DOSES DUE TO ACCIDENTS*

BYRON STATION

Postulated Accident	FSAR Section	DOSE (2 HOURS) AT EXCLUSION AREA BOUNDARY (445 meters) Thyroid (rem)	DOSE (2 HOURS) AT EXCLUSION AREA BOUNDARY (445 meters) Whole Body (rem)	DOSE (COURSE OF ACCIDENT) AT LOW POPULATION ZONE (4848 meters) Thyroid (rem)	DOSE (COURSE OF ACCIDENT) AT LOW POPULATION ZONE (4848 meters) Whole Body (rem)
Steam Line Break	15.1.5				
Conservative		2.89(+1)	4.29(-1)	4.22(+0)	2.81(-2)
Realistic		1.23(-5)	5.30(-8)	6.14(-7)	3.22(-9)
Locked Rotor	15.3.3				
Conservative		9.95(-2)	7.20(-4)	1.84(-2)	9.63(-5)
Realistic		2.98(-5)	6.09(-8)	1.98(-6)	3.90(-9)
Rod Ejection	15.4.8				
Conservative		4.43(+1)	4.19(-1)	3.90(+0)	1.66(-2)
Realistic		1.97(-5)	3.56(-8)	3.75(-6)	5.00(-9)
Steam Generator Tube Rupture	15.6.3				
Conservative		2.16(+1)	2.62(-1)	8.26(-1)	8.12(-3)
Realistic		9.25(-7)	6.21(-4)	2.16(-8)	1.45(-5)
LOCA Containment Leak	15.6.5				
Conservative		1.14(+2)	5.26(+0)	1.58(+1)	3.42(-1)
Realistic		4.04(-6)	9.23(-8)	4.42(-7)	6.85(-9)
ESF Equip. Leakage					
Conservative		7.75(-1)	2.02(-3)	1.62(-1)	1.87(-4)
Realistic		3.02(-8)	8.66(-11)	1.47(-8)	1.49(-11)

TABLE 15.0-12

POTENTIAL OFFSITE DOSES DUE TO ACCIDENTS*

BRAIDWOOD STATION

		DOSE (2 HOURS) AT EXCLUSION AREA BOUNDARY (485 meters)		DOSE (COURSE OF ACCIDENT) AT LOW POPULATION ZONE (1811 meters)	
Postulated Accident	FSAR Section	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Steam Line Break	15.1.5				
Conservative		3.90(+1)	5.79(-1)	1.76(+1)	1.17(-1)
Realistic		1.35(-5)	5.79(-8)	2.70(-6)	1.42(-8)
Locked Rotor	15.3.3				
Conservative		1.34(-1)	9.72(-4)	7.69(-2)	4.03(-4)
Realistic		3.25(-5)	6.66(-8)	8.73(-6)	1.72(-8)
Rod Ejection	15.4.8				
Conservative		5.99(+1)	5.66(-1)	1.79(+1)	7.07(-2)
Realistic		2.16(-5)	3.89(-8)	1.84(-5)	2.39(-8)
Steam Generator Tube Rupture	15.6.3				
Conservative		2.91(+1)	3.54(-1)	3.45(+0)	3.39(-3)
Realistic		1.01(-6)	6.78(-4)	9.50(-8)	6.37(-5)
LOCA	15.6.5				
Containment Leak					
Conservative		1.54(+2)	7.10(+0)	7.32(+1)	1.46(+0)
Realistic		4.41(-6)	1.01(-7)	2.16(-6)	3.24(-8)
ESF Equip. Leakage					
Conservative		1.05(+0)	2.73(-3)	7.91(-1)	8.44(-4)
Realistic		3.30(-8)	9.46(-11)	8.02(-8)	7.28(-11)

Plant characteristics and initial conditions are further discussed in Subsection 15.0.3. With three loops operating, the maximum power level (including errors) allowed in that mode of operation is assumed.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 30 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure responses shown in Figures 15.3-18 and 15.3-22 are the responses at the point in the reactor coolant system having the maximum pressure.

For a conservative analysis of thermal behavior, the hot spot evaluation assumes that DNB occurs at the initiation of the transient and continues throughout the transient. Although no rods are predicted to be in DNB, this assumption reduces heat transfer to the coolant and results in conservatively high hot spot temperatures.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reached 87% of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are full open at 2575 psia and their capacity for steam relief is as described in Section 5.4.

Evaluation of Hot Spot Temperature in the Core During the Accident

Although no rods are predicted to be in DNB, for conservatism for this accident, DNB is assumed to occur at the hot spot in the core, and therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 3.0 times the average rod power (i.e., $F_Q = 3.0$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident, although as noted above, no rods were predicted to be in DNB.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus, the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left[\frac{-45,500}{1.986\ T}\right]$$

where:

w = amount reacted, mg/cm2

t = time, sec

T = temperature, °F

The reaction heat is 1510 cal/gm.

The effect of zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Subsection 15.0.8 and listed in Table 15.0-7. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Locked Rotor with Four Loops Operating

Transient results for this case are shown in Figures 15.3-17 through 15.3-20. The results of these calculations are also summarized in Table 15.3-2. The peak reactor coolant system pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700° F. It should be noted that although no rods were predicted to be in DNB for this analysis, the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

Locked Rotor with Three Loops Operating

Bounding transient results for this case are shown in Figures 15.3-21 through 15.3-24. The results of these calculations are also summarized in Table 15.3-2. The peak reactor coolant system pressure is slightly higher than for the previous case, but is still less than that which would cause stresses to exceed the faulted condition stress limits. The clad temperature transient is less severe than for the previous case.

The calculated sequence of events for the two cases analyzed is shown in Table 15.3-1. Figures 15.3-17 and 15.3-21 show the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated seizure of a reactor coolant pump rotor (Locked Rotor Accident-LRA) assumes that the reactor has been operating with a small percent of defective fuel and leaking steam generator tubes for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant.

As a result of the accident, radionuclides carried by the primary coolant to the steam generators, via the leaking tubes, are released to the environment via the steam line safety or power operated relief valves.

The major assumptions and parameters used in the analysis are itemized in Table 15.3-3.

15.3.3.3.1 Source Term

The concentration of nuclides in the primary and secondary system prior to and following the accident are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.

   1. Accident Initiated Spike - The reactor trip associated with the LRA creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1 μCi/gm of Dose Equivalent (D.E.) I-131. The duration of the spike is assumed to be 2.5 hours.

   2. Preaccident Spike - A reactor transient has occurred prior to the LRA and has raised the primary coolant iodine concentration to 60 μCi/gm of Dose Equivalent I-131.

b. The noble gas concentrations in the primary coolant are based on 1 percent defective fuel.

c. The secondary coolant activity is based on the D.E. of 0.1 μCi/gm of I-131.

15.3.3.4 Conclusions

a. Since the peak reactor coolant system pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

b. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700° F, the core will remain in place and intact with no loss of core cooling capability. Note that for conservatism, this evaluation assumed DNB to occur at the initiation of the transient and continues throughout the transient, although no rods were predicted to be in DNB.

c. The radioactivity released to the environment as the result of a postulated LRA is presented in Table 15.3-4. The resulting thyroid and whole body doses at the exclusion area boundary and at the low-population zone outer boundary are presented in Tables 15.0-11 and 15.0-12.

15.3.3.5 Locked Rotor With a Concurrent Power Operated Relief Valve (PORV) Failure

A locked rotor event with a concurrent PORV failure was also evaluated. In evaluating the radiological consequences of this event, water level in the affected steam generator is assumed to be lost and hence, no credit for iodine partitioning is taken. The consequences for this event are bounded by the steam line break consequences presented in Section 15.1.5.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, such as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip is initiated on a low flow signal in the affected loop.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core and reduced heat transfer in the steam generators cause an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

TABLE 15.3-3

ASSUMPTIONS USED FOR THE LOCKED ROTOR ACCIDENT

	EXPECTED	DESIGN
Power	3565 MWt	3565 MWt
Fraction of Fuel with Defects	0.0012*	See Subsection 15.3.3.3
Reactor Coolant Activity Prior to Accident	ANS-N237	See Subsection 15.3.3.3
Secondary Coolant Activity Prior to Accident	ANS-N237	See Table 15.0-9
Total Steam Generator Tube Leak Rate During Accident and Initial 8 Hours	0.009 gpm	1 gpm
Activity Released to Reactor Coolant from Failed Fuel		
Noble Gas	0.0% of core inventory	0.0% of core inventory
Iodine	0.0% of core inventory	0.0% of core inventory
Iodine Partition Factor Prior to the Accident	0.0	0.01
Duration of Plant Cooldown by Secondary System After Accident, hr.	8	8
Steam Release from 4 Steam Generators	**	561,979 lb (0-2 hr) 936,100 lb (2-8 hr)
Feedwater Flow to 4 Steam Generators	793,091 lb (0-2 hr) 1,024,438 lb (2-8 hr)	793,091 lb (0-2 hr) 1,024,438 lb (2-8 hr)
Offsite Power	Available	Lost

*Per ANS-N237, American National Standard Source Term Specification.

**Condenser available, steam released through condenser off-gas system at 60 SCFM.

TABLE 15.3-4

ACTIVITY RELEASES TO ATMOSPHERE FROM LOCKED ROTOR ACCIDENT

	REALISTIC ANALYSIS ACTIVITY RELEASE (Ci)		CONSERVATIVE ANALYSES RELEASES (Ci)			
			PREACCIDENT IODINE SPIKE		ACCIDENT INITIATED IODINE SPIKE	
Isotope	0-2 Hr	2-8 Hr	0-2 Hr	2-8 Hr	0-2 Hr	2-8 Hr
I-131	6.8 (-4)	1.3 (-3)	2.5 (-1)	9.7 (-1)	2.0 (-1)	1.0 (+0)
I-132	8.9 (-5)	7.2 (-5)	6.6 (-1)	8.0 (-1)	9.1 (-1)	4.3 (+0)
I-133	7.1 (-4)	1.3 (-3)	3.9 (-1)	1.3 (+0)	3.3 (-1)	1.9 (+0)
I-134	9.2 (-6)	2.9 (-6)	2.8 (-2)	7.5 (-3)	6.1 (-2)	1.3 (-1)
I-135	2.2 (-4)	3.6 (-4)	2.0 (-1)	5.1 (-1)	2.0 (-1)	1.2 (+0)
Xe-133	1.9 (-2)	5.7 (-2)	1.71 (+1)	5.1 (+1)		
Xe-133m	3.9 (-4)	1.1 (-3)	3.5 (-1)	9.9 (-1)		
Xe-135	1.1 (-3)	2.3 (-3)	9.9 (-1)	2.1 (+0)		
XE-135m	9.9 (-6)	NEGLIGIBLE	8.9 (-3)	---	Xe and Kr Isotopes Same As PreAccident Spike Case	
XE-138	3.6 (-5)	NEGLIGIBLE	3.2 (-2)	---		
Kr-85	2.7 (-5)	8.1 (-5)	2.4 (-2)	7.3 (-2)		
Kr-85m	3.3 (-4)	5.5 (-4)	3.0 (-1)	5.0 (-1)		
Kr-87	1.4 (-4)	6.9 (-5)	1.3 (-1)	6.2 (-2)		
Kr-88	5.9 (-4)	7.0 (-4)	5.3 (-1)	6.3 (-1)		

Note: 6.8(-4) = 6.8 x 10^{-4}