Jersey Central Power & Light Company

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MEMBER OF THE General Public Utilities Corporation SYSTEM

May 23, 1975

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U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

Mr. A. Giambusso, Director
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Giambusso:

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION
DOCKET NO. 50-219
CYCLE 5 RELOAD - ADDITIONAL INFORMATION

In response to verbal concerns expressed by members of your Staff during the course of the review of the subject reload, we are submitting the responses to questions concerning the linear heat generation rate which is experienced during the rod withdrawal error transient and the fuel misloading error and its relationship to fuel cladding strain limits discussed in our reload information submittal. The linear heat generation rate corresponding to centerline melt as a function of burnup is also included.

In addition, responses to questions concerning the derivation of the overpower ratio with respect to the fuel cladding integrity safety limit, details of single channel MCHFR/MCPR calculations for various power levels and a modification of our response to question 65 regarding additional pipe break locations are included as well.

Very truly yours,

Ivan R. Finfrock, Jr.
Vice President

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Enclosures

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Are plastic strain limits exceeded in the rod withdrawal error and fuel misloading error? What is the LHGR associated with these occurrences at their worst?

Answer

Figure D-5 (page D-7) of Supplement No. 1 to Amendment No. 76 shows the peak LHGR and relative core power versus transient control rod positions. A rod block would occur at notch 14 (3.5 feet out). The peak LHGR (KW/ft) at this rod position is approximately 21.5 for the peak 7 x 7 bundle and 19.0 for the peak 8 x 8 bundle. If no rod block protection is assumed, the peak LHGR is approximately 22.0 KW/ft for the peak 7 x 7 bundle and 20.5 for the peak 8 x 8 bundle. These transient LHGR values in Figure D-5 would occur for fuel at an exposure of 3 to 6 GWD/MTM. Higher or lower exposures would result in a reduced LHGR peak for each fuel type.

The above results can be compared with the LHGR as a function of burnup for 0.75% clad strain for Oyster Creek fuel as shown in the following table:

Achieved Burnup (GWD/MTM) @ 0.75% Strain	Steady State LHGR (KW/ft)	
	8 x 8	7 x 7
30	17.5	21.5
28	18	22
22	20	24
18	21	25
15	22	26
12	23	27

It can be seen that for exposures of 18 GWD/MTM or less, when compared to the worst LHGR values for the Rod Withdrawal Error Transient (Figure D-5), the results are below the 0.75% strain values for both 7 x 7 and 8 x 8 fuel.

The fuel misloading error for the worst case misloading results in a 17% increase in the LHGR for 8 x 8 fuel and a 21% increase for 7 x 7 fuel. Assuming the fuel were operating at the limiting LHGR of 17.2 KW/ft for 7 x 7 and 14.5 KW/ft for 8 x 8, the resulting LHGR for the fuel misloading error would be approximately 20.8 for 7 x 7 and 17.0 for 8 x 8. This would not result in clad strain limits being exceeded as can be seen from the table above.

What is the LHGR at centerline melt as a function of burnup?

Answer

The LHGR at centerline melt as a function of burnup for both 7 x 7 and 8 x 8 fuel is presented in the following table:

Exposure (GWD/MTM)	LHGR @ Centerline Melt (KW/ft)
	8 x 8
0	25
1	25
2	25
4	24
10	23
15	22
25	21
	7 x 7
1	26
2	26
4	25
10	24
15	24
20	23
27.5	22

QUESTION

Discuss the derivation of the overpower ratio of 1.236 and justify the values.

RESPONSE

As discussed in the Oyster Creek Station Technical Specifications, page 2.3-3, the APRM high neutron flux scram setting has been set to assure never reaching the fuel cladding integrity safety limit. The system responds to neutron flux and is set at 120% of rated power to provide the protection while providing enough margin to rated power to avoid spurious trips. (See T.S. basis page 2.3-3, last paragraph.)

When power increases to 1690 and 1930 MWt were authorized, additional scram functions (turbine trip and generator load rejection) were added to the protective system to provide earlier response to anticipated transients which result in rapid neutron flux increases. The APRM high neutron flux scram was retained for protection against transients that result in slow power rises. For these slow maneuvers or transients, core thermal power, surface heat flux and power transferred to the coolant follow the neutron flux so a scram occurring at a neutron flux of 120% will assure thermal power has not exceeded 120% of rated thermal power. Therefore, a neutron flux scram at the safety limit would be adequate to prevent violation of the safety limit. A 3% margin between the scram setpoint and the safety limit is maintained to account for any uncertainties that might exist. This is not a derived instrument uncertainty but rather a margin included for conservatism.

Taking this information into account, a maximum steady state operating power level [illegible] be derived. Operation at no greater than this power level will assure [illegible] the safety limit is not violated for these slow power level increase transients. This power level is derived as follows: The critical power for the limiting steady state power shape is calculated. The safety limit must then be established at a thermal power which is a ratio of 1.4 below that critical power. The safety limit power is then reduced by a factor of 1.03 to provide a margin of conservatism. The resultant power is further reduced by a factor of 1.2 as discussed above in order to achieve the steady state operating power level. The ratio of the safety limit power to operating power is 1.236.

DETAILS FOR SINGLE CHANNEL

MCHFR/MCPR CALCULATIONS FOR OYSTER CREEK 8x8 FUEL

DISCUSSION

The results of a single channel MCHFR calculation for various core power levels, axial power distributions and power factors are provided herein. Four power level cases were considered. Cases I and II, 1765 MWt (MCPR = 1.73) and 3055 MWt (MCPR = 1.0), reflect radial, axial and local power factors and an axial power distribution which are characteristic of steady state power operation. Cases III and IV, 1900 MWt (MCPR = 1.40) and 2660 MWt (MCPR = 1.0), reflect the power factors and axial power distribution assumed in the evaluation of the limiting transient with respect to thermal-hydraulic limits (Rod Withdrawal ErrorTransient).

The radial, axial and local power factors for each of the respective cases are provided in Table 1 and the power distributions are provided in Table 2. The information presented in Tables 3 through 6 provides the pressure, enthalpy, mass flow, quality, mass velocity, XN-2 CHF, XN-1 CHF, rod heat flux, CHFR $\left(\frac{\text{XN-2 CHF}}{\text{Rod Heat Flux}}\right)$ and the F-factor as a function of length.

The units for each variable are as follows:

VARIABLE	ENGINEERING UNITS
Pressure	PSI
Enthalpy	BTU/lbm
Mass Flow	lbm/hr
Quality	--
Mass Velocity	10^6 lbm/hr-ft^2
XN-2 CHF	10^6 BTU/hr-ft^2
XN-1 CHF	10^6 BTU/hr-ft^2
Rod Heat Flux	10^6 BTU/hr-ft^2
CHFR	--
F-factor	--
Length	Inches

The XN-1 CHF heat flux is the heat flux calculated using the XN-1 CHF correlation with all correctors, i.e, spacer, local peaking and pressure correctors applied to the base XN-1 correlation. The XN-2 CHF heat flux is equal to the XN-1 CHF heat flux divided by the F-factor. The F-factor is defined by the following expression:

$$\text{F-factor} = \frac{C}{q''_{CHF}\,(1-e^{c\ell_{CHF}})} \int_0^{\ell_{CHF}} q''(Z)\,e^{-c(\ell_{CHF}-Z)}\,dZ$$

In the evaluation of the F-factor, the value of q'' CHF is the rod heat flux

Cases III and IV consider the values of the above variables at critical power (Pc), and at the power Pc/1.4 for the Rod Withdrawal Error Transient. Cases I and II consider the variables at the steady state operating power $\frac{Pc}{1.73}$ and the critical power, Pc, corresponding to the steady state power factors and axial power distribution given in Tables 1 and 2.

The value of the pressure for Cases I and II is conservatively higher than normal operating pressure. The pressure for Cases III and IV reflect the normal operating pressure at which the transient is initiated.

The assembly mass flow presented for each case includes allowance for 10% bypass flow and an engineering factor of 1.043 as discussed in Sections III.C.5 and III.C.6 of XN-74-32 (Rev. 3) in Amendment No. 76.

The peaking factors used in Cases I and II are discussed in Appendix C of Supplement 3 to Amendment No. 76 and are conservative yet realistic representations of actual plant operating experience.

TABLE 1

POWER FACTORS

	I*	II**
Local Peaking Factor	1.25	1.26
Radial Peaking Factor	1.485	1.68
Axial Peaking Factor	1.50	1.60
Heat Generation in Rod	.967	.967

* Cases I and II
** Cases III and IV

TABLE 2

AXIAL POWER DISTRIBUTIONS

NODE	ACTIVE LENGTH, INCHES	PEAK TO AVERAGE I*	PEAK TO AVERAGE II**
12 (top)	138	0.5	0.818
11	126	0.7	1.403
10	114	0.92	1.601
9	102	1.12	1.454
8	90	1.32	1.344
7	78	1.47	1.241
6	66	1.47	1.042
5	54	1.32	0.955
4	42	1.12	0.866
3	30	0.92	0.619
2	18	0.70	0.420
1 (bottom)	6	0.50	0.238

* Cases I and II
** Cases III and IV

TABLE 3

CASE I, 1765 MWt, MCFR = 1.73

Position	Pressure	Enthalpy	Mass Flow	Quality	Mass Velocity	XN-2 CHF	XN-1 CHF	Rod Flux	XN-2 CHF / Rod Flux CHFR	F-Factor Axial Correction
11.41	1025.527	517.95	97915.28	-.04407	.93500	1.03934	.93158	.05211	19.94467	.89632
17.12	1025.339	519.76		-.04123		1.00943		.07451	13.54729	.92288
22.83	1025.151	522.25		-.03733		.99004		.09610	10.10169	.94095
28.53	1024.840	525.42		-.03234		.98118		.11744	8.35466	.94945
34.24	1024.652	529.32		-.02625		.97956		.14000	6.96692	.95102
39.95	1024.465	533.97		-.01901		.98091		.16735	5.86125	.94971
45.65	1024.154	539.48		-.01039		.98038		.19981	4.90644	.95022
51.36	1023.968	545.93		-.00038		.96296		.22105	4.31729	.96741
57.07	1023.749	552.94		.01054		.94840		.21464	4.44511	.98226
62.77	1023.519	560.29		.02197		.94329		.24120	3.90993	.98758
68.48	1023.103	567.98		.03396		.94722	.93158	.24965	3.79419	.98349
74.19	1022.863	575.98		.04640		.94472	.91678	.26427	3.57486	.97042
79.89	1022.612	584.48		.05960		.93843	.89576	.28823	3.25581	.95453
89.60	1022.057	593.72		.07402		.91686	.87289	.31200	2.93864	.95204
91.31	1021.788	603.60		.08938		.88062	.84844	.32627	2.69905	.96346
97.01	1021.488	613.87		.10532		.84919	.82304	.33064	2.52253	.96920
102.72	1021.187	624.53		.12187		.82398	.79666	.34751	2.37108	.96684
108.43	1020.447	635.60		.13914		.80430	.76927	.36202	2.22167	[illegible]
114.13	1020.108	647.11		.15700		.79091	.74080	.38345	2.06268	[illegible]
119.84	1019.751	659.15		.17570		.76257	.71100	.40077	1.90278	.93237
125.55	1018.808	671.60		.19511		.70842	.68020	.39672	1.78568	.96017
131.25	1018.427	683.48		.21355		.63497	.65080	.36878	1.72180	1.02493
136.96	1018.040	694.53		.23070		.54219	.62345	.31658	1.71264	1.14988
142.67	1016.970	703.67		.24499		.43184	.60086	.24416	1.76871	1.39138
143.37	1016.592	710.60	97915.28	.25577	.93500	.30481	.58369	.15993	1.90591	1.91495

TABLE 4

CASE IV, 3055 MWt, MCPR = 1.0

ition	Pressure	Enthalpy	Mass Flow	Quality	Mass Velocity	XN-2 CHF	XN-1 CHF	Rod Flux	XN-2 Rod Flux CHFR	F-Factor Axial Correction
[illegible].41	1025.527	518.[illegible]1	97915.28	-.04367	.93500	1.03968	.93158	.07296	14.25089	.89602
[illegible].12	1025.339	520.74	↑	-.03970	↑	1.01041		.10432	9.68604	.92198
[illegible].83	1025.151	524.23		-.03426		.99153		.13455	7.36943	.93954
[illegible].53	1024.840	528.67		-.02731		.98323		.16442	5.98009	.94747
[illegible].24	1024.652	534.13		-.01881		.98239		.19684	4.99079	.94827
[illegible].95	1024.465	540.63		-.00869		.98508		.23430	4.20441	.94569
[illegible].65	1024.149	548.36		.00335		.98640		.27974	3.52612	.94443
[illegible].36	1023.905	557.38		.01737		.96907		.31227	3.10334	.96131
[illegible].07	1023.657	567.20		.03263		.95309		.32822	2.90382	.97743
[illegible].77	1023.401	577.49		.04862		.92814	.91304	.33776	2.74795	.98373
[illegible].48	1022.883	588.25		.06538		.90552	.88643	.34951	2.59082	.97892
[illegible].19	1022.601	599.45		.08279		.89206	.85870	.36997	2.41114	.96260
[illegible].89	1022.304	611.35		.10126		.88262	.82927	.40353	2.18728	.93955
[illegible].60	1021.592	624.28		.12142		.85757	.79726	.43680	1.96329	.92968
[illegible].31	1021.255	638.12		.14289		.81469	.76303	.45678	1.78[illegible]56	.93659
[illegible].01	1020.894	652.50		.16520		.77486	.72746	.47130	1.64410	.9388[illegible]
[illegible].72	1020.513	667.42		.18835		.74185	.69054	.48652	1.52482	.930[illegible]
[illegible].43	1019.531	682.92		.21249		.71688	.65219	.50683	1.41444	.90975
[illegible].13	1019.113	699.03		.23747		.70217	.61233	.53683	1.30801	.87205
[illegible].84	1018.674	715.89		.26361		.67820	.57061	.56108	1.20875	.84136
[illegible].55	1017.420	733.32		.29073		.62727	.52749	.55541	1.12937	.84094
[illegible].25	1016.935	749.95		.31651		.55244	.48633	.51629	1.07001	.88034
[illegible].96	1016.432	765.43	↓	.34049	↓	.45475	.44805	.44321	1.02602	.98527
[illegible].67	1014.971	778.21		.36043		.34139	.41641	.34182	.99874	1.21975
[illegible].37	1014.471	787.93	97915.28	.37349	.93500	.22165	.39238	.22390	.98996	1.77028

TABLE 5

CASE III, 1900 MWt, MCPR = 1.40

Position	Pressure	Enthalpy	Mass Flow	Quality	Mass Velo	XN-2 CHF	XN-1 CHF	Rod Flux	XN-2 Rod flux / CHFR	F-factor Axial Corrector
11.41	1240.292	518.69	95205.06	.09665	.90912	.97466	.93466	.10070	9.67929	.95696
17.12	1240.104	521.97		.09138		.95790	.93466	.11991	7.98901	.97570
22.83	1239.417	525.92		.08481		.95446	.93466	.13580	7.02852	.97926
28.53	1239.598	530.51		.07715		.95563	.93466	.15286	6.25173	.97806
34.24	1239.411	535.71		.06353		.95756	.93466	.17267	5.54508	.97609
39.45	1239.224	541.53		.05388		.95838	.93466	.19470	4.92231	.97526
45.65	1238.904	508.04		.04306		.95751	.93466	.21710	4.00923	.97614
51.36	1238.721	555.26		.03510		.95301	.93466	.23718	4.01875	.98075
57.07	1238.536	563.08		.02315		.94604	.93466	.25010	3.79062	.98589
62.77	1238.351	571.33		.00949		.94516	.93144	.26207	3.60651	.95868
66.48	1238.036	580.01		.00490		.92983	.91039	.27425	3.39043	.97909
74.19	1237.810	584.12		.01490		.91146	.88827	.29407	3.09945	.97455
79.89	1237.573	598.75		.03592		.87948	.86491	.30669	2.86768	.98343
89.60	1237.118	608.61		.05229		.83610	.84093	.30267	2.76241	1.00582
91.31	1236.871	618.13		.06605		.79641	.81786	.28814	2.76396	1.02693
97.01	1236.617	627.31		.08323		.76502	.79561	.27061	2.82707	1.03998
102.72	1236.359	635.45		.09753		.73666	.77164	.25303	2.91131	1.05155
108.43	1235.792	644.01		.11094		.70650	.75507	.23404	3.01823	1.06876
114.13	1235.529	651.46		.12326		.67390	.73701	.21214	3.17665	1.09364
119.84	1235.266	658.19		.13441		.64379	.72068	.19004	3.38767	1.11900
125.55	1234.626	654.20		.14444		.61707	.70608	.18910	3.68427	1.14423
131.25	1234.366	669.59		.15337		.59414	.69301	.15008	3.95875	1.16640
136.96	1234.103	674.37		.16129		.57320	.68141	.13279	4.31600	1.18877
142.67	1235.419	676.53		.16827		.54845	.67132	.11585	4.74243	1.22403
148.37	1233.154	682.03	95205.06	.17406	.90912	.50636	.66243	.09531	5.31292	1.30900

TABLE 6

CASE IV, 2660 MWt, MCPR = 1.0

POSITION	PRESSURE	ENTHALPY	MASS FLUX	QUALITY	MASS VELO	XN-2 CHF	XN-1 CHF	ROD FLUX	XN-2 Rod Flux CHFR	F-factor AXIAL COLLECTOR
11.41	1240.293	519.71	95205.06	.09517	.50942	.97522	.43460	.17420	5.59816	.95842
17.12	1240.105	545.34		.09575		.95900	.43460	.20740	4.62303	.97462
22.43	1239.917	532.22		.07942		.95600	.43460	.23493	4.06928	.97768
28.53	1239.598	540.15		.06123		.95420	.43460	.26445	3.62358	.97540
34.24	1239.411	549.15		.04634		.46164	.43460	.29872	3.21441	.97140
39.95	1239.224	554.21		.02449		.46334	.43460	.33643	2.44297	.96423
45.65	1238.908	570.47		.01103		.46417	.41352	.37564	2.56642	.96521
51.36	1238.713	542.97		.00903		.42991	.40320	.41029	2.26607	.97120
57.07	1238.453	596.50		.03242		.47870	.47037	.43264	2.45904	.97645
62.77	1238.1 4	610.77		.05962		.45796	.43574	.45330	1.59937	.97512
68.44	1237.661	625.78		.08350		.82772	.79931	.47445	1.74454	.96568
74.19	1237.364	641.55		.10557		.80081	.76105	.50870	1.57404	.99035
79.89	1237.054	658.21		.13411		.75680	.72063	.53057	1.42549	.99254
85.60	1236.323	675.27		.16234		.69300	.67922	.52362	1.32348	.98011
91.31	1235.790	691.74		.18961		.62887	.63425	.49849	1.26076	1.01716
97.01	1235.640	707.61		.21583		.47070	.60075	.46815	1.21426	1.05248
102.72	1235.296	722.56		.24054		.51627	.56447	.43773	1.1442[illegible]	1.08705
108.43	1234.366	726.51		.25362		.46484	.53062	.40493	1.15784	1.13166
114.13	1234.003	744.34		.24494		.41783	.49418	.36701	1.13614	1.14741
119.84	1233.036	761.03		.34418		.36782	.47112	.32871	1.11674	1.28087
125.35	1232.964	771.44		.32123		.32331	.44586	.29251	1.10922	1.37[illegible]2
131.25	1232.181	780.76		.33347		.38400	.42325	.24462	1.09362	1.49830
136.96	1231.794	789.03		.35054		.24400	.40318	.22973	1.09386	1.61926
142.47	1230.531	798.23		.36252		.21710	.38571	.20007	1.08515	1.77668
148.37	1230.238	802.28	95205.06	.37651	.50412	.17778	.37104	.18428	1.07820	2.08710

QUESTION

65. The spectrum of breaks submitted does not meet requirements of Appendix K to 10 CFR Part 50, which requires that the Moody multiplier range from 0.6 to 1.0 be spanned. It is our position that the spectrum of breaks analyzed must include recirculation line breaks with approximate areas (in square feet of 0.75, 2.0 and 4.0). In addition as per our discussions, the following break locations must be analyzed: feedwater line, main steam line and core spray line. These analyses must be performed for both GE and Exxon fuel.

RESPONSE

The results of the blowdown and heatup analyses for GE fuel in Oyster Creek for the 0.75 ft^2, 2.0 ft^2, 4.0 ft^2, main steamline, feedwater line and core spray line breaks are included in the following figures. In addition, we have included the peak cladding temperature and heat transfer coefficients versus time for the 0.02 and 0.05 ft^2 breaks which were omitted from our April 24, 1975 submittal because of their proprietary nature.

The core spray system at the Oyster Creek Nuclear Generating Station consists of two identical loops either capable of supplying rated core spray flow. The active components in each loop are redundant, each loop is powered by a separate diesel generator and the two loops are completely separated from each other so that effects on one loop from missiles or rupture will not affect the other loop. The system is therefore designed to deliver rated core spray flow to the reactor vessel in the event of an unlikely loss-of-coolant accident even if there is no off-site power available and emergency power is not available to one of the core spray systems due to a fault on one of the emergency diesel buses. This is the case with all postulated pipe breaks within the reactor coolant system pressure boundary except for a break in one of the core spray lines between the reactor vessel and the core spray check valves, a run of approximately 28 ft. of 6 inch ID pipe in each of the two core spray loops. Should the loss-of-coolant accident result from a break in this portion of a core spray loop, and no off-site power is available and the diesel generator bus which powers the other core spray loop is assumed to be the single pressure failure, core spray flow would not automatically reach the core. However, the system is designed to indicate this event to the operator and sufficient time is available for the operator to supply water to the core to permit cooling sufficient to meet the NRC's Final Acceptance Criteria for ECCS.

The reliance on operator action in the emergency core cooling sequence in this case is justified for several reasons:

1. The specific event probability is very small. The probability of any loss of coolant accident is small in itself and the probability of a rupture in a specific run of pipe the length of which is small compared to the total pipe in the reactor coolant pressure boundary is even smaller. In all likelihood, were a LOCA to occur, emergency

loads including core spray would be powered from the highly reliable off-site power network. The low probability of its unavailability and the simultaneous failure of the diesel generator associated with the core spray system which is still intact when coupled with the very low probability of the LOCA in the core spray line results in a very unlikely event.

2. The system is designed to alert the operator to this specific event.

In addition to all of the normal indications of a LOCA (i.e., reactor scram, high drywell pressure, decreasing water level, containment isolation, etc.) the operator is provided with a specific visual and audible alarm (individual one for each core spray loop) which reads "Core Spray System I Pipe Break" and "Core Spray System II Pipe Break". These alarms are initiated by differential pressure detectors on each core spray loop which compare the pressure in the bottom plenum of the reactor vessel with that in the core spray line just upstream of the reactor vessel nozzle. If the pressure in the core spray line drops 18 psi below that in the reactor vessel lower plenum, the alarm is initiated signaling a depressurization of the core spray line, that is, a rupture. Regardless of the location of the break in the non-isolable portions of the core spray system these sensors would sense the depressurization. The differential pressure detectors themselves are outside the primary containment and the sensing lines within containment are routed so as to preclude damage from a rupture in the other core spray system.

3. Concise procedures are available and operator action is quickly accomplished. Emergency Procedures call for verification of both core spray systems running after LOCA indications. Furthermore, indication of a "Core Spray System Pipe Break" calls for verification that the other core spray system is operable. If the second core spray system diesel is not operating then procedures call for reduction of load on the operating diesel, connecting the two emergency buses together and once this is accomplished the emergency safeguards loads on that bus, including core spray will automatically sequence on. If the failure is due to a fault on the bus, which powers the operable core spray system, the operator will not be able to close the control room breaker which interconnects the two emergency buses. In order to supply adequate cooling to the core the operator must take action to start pumping water to the core by means of a condensate pump. There exists adequate water supply for the condensate pump for at least ten minutes (45,000 gal.) from the condenser hot wells. In less than three (3) minutes the core would be covered. After seven minutes of operations the condensate pump could be secured due to the fact that the event has been turned around and the reactor pressure has been reduced to a level such that the fire pond pumps can supply adequate water through the core spray header to maintain the reactor water level above the active fuel.

On May 21, 1975 the procedure to initiate the condensate pump water to the core was walked through at Oyster Creek and the operator was able to effect the necessary actions in less than six (6) minutes.

Even recognizing the pressure of the emergency situation, this demonstration, the availability of clear indication of the problem, and concise action requirements, and the fact that all immediate actions required to establish adequate cooling flow can be accomplished provides reasonable assurance that these actions can be accomplished in time to provide adequate cooling required to meet the ECCS Final Acceptance Criteria.

RESULTS OF CORE SPRAY LINE BREAK
LOSS-OF-COOLANT ACCIDENT EVALUATION

ACCIDENT: Break in one loop of core spray system.

SINGLE FAILURE: Fault in bus which provides power to unbroken core spray loop.

MITIGATION ASSUMPTIONS: Condensate pumps initiated 562 seconds into event.

HEATUP RESULTS (EXXON NUCLEAR FUEL)

Time (Sec)	PCT (^{o}F)	HTC
0	503	19,306
2.2	571	19,306
2.3	574	2,467
5.0	926	46
7.0	1028	44
9.0	1087	35
13.4	1136	34
20.5	1141	34
27.0	1130	34
35.5	1113	34
47.0	1089	34
62.5	1053	34
82.5	968	42
95	936	42
125.5	887	38
161.5	863	32
191.0	868	25
219.5	887	20
252.5	919	16
290.5	974	12
310.0	990	13
310.5	996	0
384.0	1347	0
510.0	1804	0
590.0	2139	0
595.0	2021	20
675.0	1221	20
775.0	842	20
890.0	687	20

% Local Metal-water
Reaction = 2.23%

SENSITIVITY TO Tpump

Tpump (sec)	PCT (^{o}F)	% MWR
400	1481	0.16
500	1824	0.74
562	2139	2.28
600	2425	4.91