TERA
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Three Mile Island

APR - 7 1979

MEMORANDUM FOR: Brian Grimes, Assistant Director for Engineering Projects, DOR

THRU: S. H. Hanauer, Assistant Director for Plant Systems, DSS

FROM: R. M. Satterfield, Chief, Instrumentation and Control Systems Branch, DSS

SUBJECT: SUMMARY OF RECENT STAFF EFFORTS TO ASSESS ENVIRONMENTAL QUALIFICATION OF SAFETY RELATED ELECTRICAL EQUIPMENT INSTALLED AT TMI-2

In response to the recent event at TMI-2, an effort was initiated by DSS and DOR personnel to establish the radiation withstand capability of specific pieces of safety related electrical equipment installed at TMI-2. This information, together with calculated radiation levels inside of containment, was to be used to assess the possibility of failure of this equipment due to radiation exposure as a function of time. This memorandum summarizes the status of this effort. Work thus far indicates that time of failure is difficult to establish because of uncertainties in the calculated dose rates at various locations inside containment. Better predictions would be possible only when better data become available on which to base dose calculations.

1. <u>Qualification Data for Selected Safety Related Electrical Equipment Installed AT TMI-2</u>

   1. Instrumentation Related to Coolant Temperature, Pressure and Flow Measurements

Table I lists those pressure and differential pressure transmitters and temperature sensors located inside containment which are considered important to the current mode of operation (Reactor coolant pump operating and providing

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flow through one steam generator.) This table also presents the elevation of the transmitters above the containment floor (282' 6" level). All instruments are located in the annular region between the shielding wall and the containment wall. The table also shows the radiation dose for which the transmitter was designed and/or tested.

In an attempt to gather more information on those transmitters suspected to be particularly susceptable to radiation damage (i.e. both the Bailey and Foxboro transmitters), DOE-Naval Reactors and Sandia Laboratories were asked to search their files for radiation test data on these instruments. While neither organization had made use of the specific model instruments of concern, Sandia Laboratories discovered through discussions with the equipment manufacturer that certain Foxboro pressure transmitters installed at TMI-2 are equipped with radiation hardened amplifiers and that these transmitters have survived doses as high as 2×10^8 rads. These instruments are used to measure reactor coolant pump seal cavity pressure at TMI-2. (The identification numbers for these instruments are RC22-PT4 through -PT8). Should existing reactor coolant pressure transmitters fail, these hardened transmitters may continue to be available to measure reactor coolant pressure.

1.2 Allis Chalmers Reactor Coolant Pump Motors

B&W has established that the surge capacitors are radiation sensive. They estimate that capacitor failure is expected to occur between dose levels of 3.1×10^6 and 4.2×10^6 Rads. The mode of failure is thermal runaway. This will lead to short-circuitry of the capacitor which will trip the supply circuit breaker. If the capacitor remains short-circuited, the motor cannot be restarted or operated. If the capacitor should open-circuit, the motor can be started and operated without it. Therefore, it appears worthwhile to attempt a restart in the hope that either the capacitor has opened up or that the starting pump will open it up.

1.3 Pressurizer Vent, Block and Relief Valves

With regard to the block and relief valves, B&W estimates that the design radiation level is 2×10^8 rads. This value was based on tests performed on similar valves.

The pressurizer vent valves were supplied by Burns and Poe and that organization has been contacted to establish the appropriate design radiation level for those valves.

1.4 36" Recombiner Isolation Valves

These valves were supposedly qualified to 2×10^7 Rads. However, on April 6, 1979, Region III notified NRR that the valve manufacturer had filed a Part 21 notification indicating that at least one component of the valve had been tested only to 4×10^5 Rads. As noted below, the effect of this reduced design radiation level has yet to be assessed. This issue is being pursued with the manufacturer.

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1.5 Fan Cooler Motors

It was established that the fan cooler motors survived exposure to 10^9 Rads.

1.6 Decay Heat Removal System Components

Components of the decay heat removal system which may be affected by radiation have been identified. However, design radiation levels have yet to be established. In view of current plans not to use the DHR system, there are no efforts underway to develop this information.

2. Estimates of Time of Failure of Safety Related Equipment Due to Radiation Exposure

In an attempt to predict when critical electrical equipment installed at TMI-2 might fail due to radiation exposure, dose rates calculated for various points inside containment were used in conjunction with the design radiation levels described in the previous section to establish equipment lifetime.

The e rate calculations were based on analyses of containment air and coolant samples taken on March 31, 1979. (See Enclosure 2) It was assumed that the air sample was representative of the average containment atmosphere and that the coolant sample represented the liquid on the containment floor. It was further assumed that the contributions to the calculated dose due to radiation from the reactor vessel, steam generators and pressurizer were negligible, because of the shielding between these components and the equipment of concern.

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This latter assumption was supported by calculations which showed that little of the radiation produced by coolant within these vessels penetrated the vessel walls.

2.1 ORNL Estimate of Radiation Dose at Location of Foxboro and Bailey Transmitters.

ORNL, using a two dimensional computer calculation, estimated the dose rate delivered to those Foxboro and Bailey transmitters that are a part of the reactor coolant pressure and flow measure system. (See Table 1) These transmitters are located in the annular region between the containment wall and the shielding wall. They are installed on an instrument rack and are positioned several feet (See Table 1) above the containment floor. The dose to these instruments is due to radiation from both the containment atmosphere and the contaminated liquid covering the containment floor.

One cause for inaccuracy of the dose rate estimates is the assumption that the coolant sample taken on March 31 is respresentative of the water on the containment floor. This sample was taken several days after the event whereas a significant portion of the water on the containment floor was deposited during the early stages of the event and should, therefore, be much less contaminated than the March 31 sample.

From these calculations it was estimated that the transmitters are exposed to a dose rate of 1×10^4 Rads/hr. The Bailey Model BY pressure transmitter was tested to withstand 5×10^4 Rads with no effect on output and 4×10^7 Rad with slight shift in calibration of 4% (BAW 10003). Several of these transmitters apparently continue to function normally. It is therefore assumed that their capability above 5×10^4 Rad is being used.

2.2 Calculated Doses to Other Components Inside Containment

Other calculations were performed by the NRC staff to estimate the dose received by certain other important components inside containment. In these calculations it was assumed that the only isotope of importance in the containment atmosphere is Xe^{133}. The analysis of the containment air sample showed Xe^{133} to be present in the concentration of 675 microcuries/cm^3 (Table II). To be conservative and to account for additional Xe^{133} being vented into the containment from the pressurizer as a part of the degassing of the primary coolant, the calculations were performed assuming a Xe^{133} concentration of 1000 microcuries/cc.

The immersion dose due to Xe^{133} was estimated as a function of room size. The room enclosure was assumed to be spherical and to be bounded by a significant amount of shielding for the soft gamma (0.08 Mev) associated with Xe^{133} decay. Dose rates at various locations in the containment

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were approximated by estimating a "charac. ristic radius" for the enclosed area of interest using the best available building drawings. In addition to this air dose from Xe^{133}, the dose from I^{131} dissolved in the water on the containment floor was also calculated where applicable. This calculation was performed using the concentration measured in the primary coolant sample (Table II). These calculations are entirely dependent on the radioactivity concentrations assumed in the air and water. Uncertainties in these values would introduce corresponding uncertainties into the calculational results.

These calculations produce radiation dose rates as follows:

Various valves located on top of the pressurizer	800 Rad/hr
Reactor Coolant pump surge capacitors	1100 Rad/hr*
Electrical penetrations	1500 Rad/hr

Based on current knowledge of the design radiation level for pressurizer block and relief valves, the lifetime is estimated to be 2.5 x 10^5 hours. The surge capacitors are calculated to last 3000 hrs. The design radiation level for the penetrations has not yet been established.

An estimate was also made of the dose delivered to the 36" Recombiner Isolation Valves. In this calculaion it was assumed that the major damage mechanism was exposure of the valve seat due to the flow of containment

*B&W estimated that the integrated dose to the surge capacitors as of April 4, 1979 was 1.8 x 10^6 Rads. They estimate that thermal runaway could begin 12 days after the event. The B&W dose calculation was based on consideration of contributions from both containment atmosphere and liquid at the bottom of the containment. The NRC calculations were based on the assumption that the capacitors would be shielded from the liquid at the bottom of containment. The difference between the NRC and B&W calculations will be investigated further.

gas passing through the valve. It is believed that the metal exterior of the valve will serve to shield valve components from the radiation from the containment atmosphere surrounding the valve.

The radiation dose to the valve seat is estimated to be about 2000 Rad/hr with most of this dose due to beta rays. The staff estimated the radiation level capability for the valve seat material for this valve to be about 10^6 Rads. Therefore, the seat material should survive at least 21 days. Assuming the seat is thick relative to the penetration capability of the beta rays, the integrity of the seat would be expected to be maintained substantially beyond that time.

As noted above, the valve manufacturer has indicated that certain valve components, particularly organic material used in a solenoid valve mounted on the isolation valve, were tested only to 4×10^5 Rads. The best information now available indicates that failure of the solenoid valve would prevent closure of the isolation valve. However, without more detailed knowledge of the solenoid valve design and the shielding afforded the organic material by the valve body, it is impossible to establish whether, and at what level, radiation will cause failure of the solenoid valve. The manufacturer will provide appropriate drawings shortly.

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TABLE 1

B&W Instruments Inside Containment

IA - INST MOUNTED BY ITSELF

IR - INST MOUNTED ON RACK

Parameter	Inst Ident #	Mounting	Type	Rad Level Design Test	Above 282.5' Level
SG "B" PRESS	SPGB-PT1	IM 13	Fox E11GH	10^7	2' 5"
RC Flow LpA	RC14A-DPT-3&4	IM-14&15	Bailey BY	10^7	3' 0"
PRZ LEVEL	RC.1-LT 1, 2&3	IR-424&425	Bailey BY	10^7	3' 6"
SG "B" PRESS	SP 6B-PT2	IR-428	Fox E11GH	10^7	
RC FLOW LpA	RC14A-DPT1&2	IR-425&426	Bailey BY	10^7	
RC FLOW LpB	RC14B-DPT3&4	IR-429&430	Bailey BY	10^7	
RC FLOW LpB	RC14B-DPT1&2	IM-12&13	Bailey BY	10^7	
SC "A" LEVEL	SP1A-LT2&3	IR-426	Bailey BY	10^7	
SG "A" LEVEL (SU)	SP1A LT4&5	IR-426	Bailey BY	10^7	
SG "B" LEVEL	SP1B LT1, 2&3	IR-428	Bailey BY	10^7	
SG "B" LEVEL (SU)	SP1B LT4&5	IR-428	Bailey BY	10^7	
SGA LEVEL	SP1A LT1	IR-426	Bailey BY	10^7	5' 2"
SG "A" PRESS	SP6A PT1&2	IR-426&424	Fox E11GH	10^7	
RC PRESS (WR)	RC3A PT3&4	IR425&427	Fox E11GH	10^7	
RC PRESS (NR)	RC3A PT5	IR424	Fox E11GH	10^7	
RC PRESS (WR)	RC3B PT3	IR429	Fox E11GH	10^7	
RC TEMP (NRT_c) (A LOOP)	RC5A TE2&4		Rosemount 177Y	10^8	
RC TEMP (NRT_c) (B LOOP)	RC5B TE2&4		Rosemount 177Y	10^8	
RC TEMP (WRT_H) (A LOOP)	RC15A TE1				
RC TEMP (WRT_H) (A LOOP)	RC15A TE2&4				
RC TEMP (WRT_H) (B LOOP)	RC15B TE1				

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TABLE 1 (Cont'd)

Parameter	Inst Ident #	Mounting	Type	Rad Level Design Test	Above 282.5 Level
RC TEMP (WRT_c) (B LOOP)	RC15B TE2&3		Rosemount 177Y	10^8	
PRZ TEMP	RC2 TE 1&2				
RCP1A SEAL RET FLOW	MU10-FT1		BROOKS		
RCP1B SEAL RET FLOW	MU10-FT3		BROOKS		
RCP2A SEAL RET FLOW	MU10-FT2		BROOKS		
RCP2B SEAL RET FLOW	MU10-FT4		BROOKS		

TABLE 1 (Cont'd)

EQUIPMENT INSIDE CONTAINMENT NECESSARY
FOR CONTINUED OPERATION IN PRESENT MODE

Component	Identification No.	Type	Design or Test & Level	Elevation Feet
Pressurizer				
Vent Valve (Mov)	RC-V117			352'
Block Valve (Mov)	RC-V2		2.04×10^8	355'
Relief Valve (SV)	RC-R2			355'
Spray Valve				355'
Heaters				321'
Reactor Building Air Cooling Units			1×10^9	
Containment Isolation Valves on Purge System				
In Core Thermocouples				
Reactor Coolant Pump Motor and Surge Capacitors				

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TABLE 2

Results of Analyses of Containment Air and Coolant Samples Taken March 31, 1979

Constituents of Contaminated Coolant

I^{133} - 1.3×10^4 microcuries/cc

I^{133} - 6.5×10^3 "

Cs^{136} - 1.8×10^2 "

Cs^{137} - 2.8×10^2 "

$Barium^{140}$ - 200 "

Constituents of Contaminated Containment Atmosphere

Xe^{133}	-	675	microcuries/cc
Xe^{m}	-	15	"
X^{135}	-	8.1	"
I^{131}	-	.063	"

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