Met-Ed / GPU

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Writer's Direct Dial Number

October 17, 1979
GQL 1294

Office of Nuclear Reactor Regulation
Attn: Harold R. Denton, Director
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Units 1 & 2 (TMI-1 & TMI-2)
License Nos. DPR-50 & DPR-73
Docket Nos. 50-289 & 50-320
Safety and Non-Safety Grade System Interaction

This letter and the attached report responds to your September 17, 1979 letter on "Potential Unreviewed Safety Question on Interaction Between Non-Safety Grade Systems and Safety Grade Systems", and is being submitted late as discussed by Mr. D. DiIanni of NRC and Mr. J. R. Stair of my staff on October 4, 1979.

Within the time frame allowed by your letter, our efforts have been focussed on developing and understanding of the issue and making a preliminary assessment of the impact of this issue on the conclusions of the safety analyses presented in the FSAR.

We have not discovered any adverse interactions which we feel warrants license modifications. There are, however, areas which will require additional investigation. These areas are detailed in the attached evaluation.

The attached report, which was prepared in conjunction with Babcock & Wilcox and Metropolitan Edison Company, provides the details of our review and identifies further actions that we are undertaking to address the longterm system response under adverse environmental conditions.

Because of the existing situation at TMI-2, the site specific part of this evaluation only addresses TMI-1.

Sincerely,

J. G. Herbein
Vice President-Nuclear Operations

JGH:LWH:tas

Attachment

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METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY
THREE MILE ISLAND NUCLEAR STATION, UNITS 1 & 2

Operating License Nos. DPR-50 and DPR-73
Docket Nos. 50-289 and 50-320

This letter is submitted in support of the Nuclear Regulatory Commission's request concerning safety and non-safety grade system interaction, dated September 17, 1979, for Three Mile Island Nuclear Station, Units 1 and 2. As a part of this request, an "Evaluation of Potential Adverse Environmental Effects on Non-Safety Grade Control Systems" is attached. Further, all statements contained in this report have been reviewed and all such statements made and matters set forth therein are true and currect to the best of my knowledge, information, and belief.

METROPOLITAN EDISON COMPANY

By J G Herbein
Vice President

Sworn and subscribed to me this 17th day of October, 1979.

L F Lawyer
Notary Public

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EVALUATION OF POTENTIALLY ADVERSE ENVIRONMENTAL EFFECTS ON NON-SAFETY GRADE CONTROL SYSTEMS

Prepared by:

METROPOLITAN EDISON COMPANY
and
BABCOCK AND WILCOX COMPANY

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October , 1979

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I. Introduction

This report is in response to Harold R. Denton's letter of September 17, 1979, on the subject of "Potential Unreviewed Safety Question on Interaction Between Non-Safety Grade Systems and Safety Grade Systems." It addresses the concerns listed in Information Notice 79-22 and fulfills the commitment made during our meeting with your staff on September 20, 1979. During that meeting, we committed to:

Evaluate the impact on the Licensing basis accident analyses due to consequential environmental effects on non-safety grade control systems.

- Identify Licensing basis accidents which cause an adverse environment for each plant.

- Define Safety Analysis inputs and responses used during Licensing basis accidents

- Verify Safety Analysis conclusions or recommend actions justifying operation.

The scope of this response includes an evaluation of the plants equipment actuation and performance and effects on the Licensing basis analysis. A matrix of potential environmental effects on non-safety grade control systems is presented. Where non-safety grade equipment performance could be affected by the adverse environment, a safety assessment will be prepared. The safety assessment will define any potential problems which may arise due to the effects of an adverse environment on non-safety grade control systems.

Work beyond the scope of the 20-day response and work to provide a more detailed assessment are included in recommended followup actions.

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II. Plant Licensing Basis

1. Safety Analysis Functions and Parameters

The plant licensing basis analyses, as presented in the FSAR, were reviewed to define the inputs, assumptions and responses used for non-safety grade control systems. This information is summarized in Table I, which lists typical equipment actions and actuation times used in the safety analyses for B&W 177 fuel assembly plants. The data has been categorized to reflect the functional requirements as follows:

A. Reactor Power Control and Shutdown

B. Reactor Pressure Control

C. Steam System Isolation and Pressure Control

D. Feedwater System Isolation and Control

This categorization has been developed to focus upon those primary functions which have a potential for control system interaction.

The table identifies the range of equipment actions and actuation times used in the plant safety analysis for steam line break, feedwater line break and large and small LOCA.

2. Plant Unique Features

A. Steam Line Pressure Switches

TMI-1 has steam line pressure switches located inside containment that actuate to isolate feedwater on a Steam Line Break (SLB). These pressure switches would be exposed to an adverse environment on a line break inside containment. Failure of a single pressure switch will not prevent isolation of feedwater nor will failure of a single pressure switch cause it to incorrectly isolate feedwater. The pressure switches have previously been evaluated as suitable for steam line break environment. This previous conclusion will be reverified.

B. Feedwater Isolation System

The feedwater isolation system for SLB was modified during the 1979 refueling outage to close the main and startup feedwater block valves on the affected steam generator in addition to the control valves. The block valves are motor operated and therefore are unlikely to inadvertently reopen after closing. Although the valves may not be qualified for adverse environment the redundancy provides additional assurance of feedwater isolation.

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C. Pressurizer Heaters

The pressurizer heater control and power supply could be exposed to adverse environment on a feedwater line break. A feedwater line break could affect pressurizer heater operation, however, this would not adversely affect the course of the accident.

D. Auxiliary Feedwater System

The main steam lines and one feedwater line go through the Intermediate building. The Auxiliary Feedwater System is also located in the Intermediate Building. The effects of adverse environment on the Auxiliary Feedwater System have not been fully evaluated. Although there is some distance separating the Auxiliary Feedwater components and themain feedwater and steam lines it is not possible to state at this time that Auxiliary Feedwater components would not be affected.

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III. Safety Assessment

1. Potential Environmental Effects

The non-safety grade control systems have been reviewed to determine if an accident environment could adversely affect the analyzed course of the event. Specifically, the approach taken was to use the safety analysis function and parameters from Table I as a basis to identify where potential control system effects could have an impact. The result of this evaluation is summarized in Table II, Potential Environmental Effects on Non-Safety Grade Control Systems. The matrix identifies, for six accident types, the non-safety grade control systems which could be adversely affected by the environment caused by the event. Where no entry is made in the matrix, no potential for environmental effects exists due to the physical location of the equipment with respect to the high energy line break, i.e., breaks inside containment do not affect equipment outside containment and vice versa. If an entry is made (X or Y), a potential effect exists as follows:

X - The adverse environment caused by the break could affect the equipment and, equipment malfunction could affect safety analysis functions identified in Table I.

Y - The adverse environment caused by the break could interact with the equipment, but the equipment malfunction would not affect safety analysis functions identified in Table I.

This structuring of the potential effects matrix provides a focus on those non-safety grade control systems which are important and identifies areas for further evaluation of the impact on the safety analysis (i.e., X's).

2. Impact on Plant Safety Analysis

Potential environmental effects which could adversely impact the plant safety analysis are identified in Table II with an "X". For each potential adverse effect, a safety assessment will be prepared to confirm plant safety or identify a potential problem area.

A. Turbine Bypass/Atmospheric Relief Valves, MFW Control and AFW Control Under Large LOCA Environment

The large break loss-of-coolant accident relies upon safety grade equipment for mitigation. The potential effects presented in Tables I and II indicate that the control system functions, though considered in the analysis, are modelled conservatively such that postulated malfunctions of these systems will not invalidate the analytical results. The reactor shutdown and pressure control during the blowdown and reflood phases do not rely upon non-safety grade control systems. The steam and feedwater system control features are conservatively modelled in the analyses as follows:

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1. The secondary steam system is conservatively assumed to remain intact (bottled up) to provide a large heat source during the late stages of blowdown. The steam safety valves are assumed to maintain a conservately high steam pressure. Potential control system effects which provide steam relief would tend to improve the analytical results.

2. The feedwater system flow is conservatively assumed to quickly decrease to zero following the break. This loss of feedwater minimizes the effect of the OTSG secondary as a heat sink for a conservative analysis.

B. MFW and AFW Control and Turbine Bypass/Atmospheric Relief Under Small LOCA Environment

The small break loss-of-coolant analysis has been revised since TMI-2 to include a parameterization of potential equipment and operator actions during the accident. As a result of this re-analysis, operating guidelines have been prepared by the NSSS vendor for use in operator training and revised operating procedures. This change to the small break operating procedures provides a consistency between the small LOCA safety analysis and the required equipment and operator actions.

A review of Table II indicates a potential problem with the main or auxiliary feedwater level control. The small break analysis and operating guidelines utilize OTSG level for RCS cooling and depressurization. In the adverse environment caused by the small LOCA, the OTSG level indication could potentially be misleading to the operator and cause an inadequate amount of OTSG water inventory.

The effect of errors due to reference leg heating are being evaluated in response to IE Bulletin 79-21.

Table II also indicates a potential interaction between small LOCA and Turbine Bypass and Atmospheric Relief Valve control. The steam generator outlet pressure transmitters are located inside containment and thus could be exposed to an adverse environment. The transmitters were procurred with environmental qualifications. It is expected that adverse interaction is unlikely; however, further investigation is planned.

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C. Pressurizer PORV under SLB (Inside Containment), FWLB (Inside Containment) and LOCA Environments

The analysis and consequences of inadvertent opening or failure to close of the pressurizer PORV as a result of SLB, FWLB or small LOCA environment will be addressed in the "TMI-1 Restart Safety Analysis Report" and will be submitted prior to TMI-1 restart.

D. CRDCS Under All Environments

Potential for a significant increase in initial power level as a result of spurious rod withdrawal prior to reactor trip for SLB, FWLB and LOCA will be addressed in the "TMI-1 Restart Safety Analysis Report" and will be submitted prior to TMI-1 restart.

E. MFW and AFW Control and AFW Isolation Valves Under SLB, FWLB; and Small LOCA Environments

As indicated in Table 1, these control systems are important elements of the safety analyses for steam and feedwater line breaks. These accidents will be addressed in the "TMI-1 Restart Safety Analysis Report" and will be submitted prior to TMI-1 restart.

IV. Justification for Operation

Based on the evaluations and safety assessments which will be presented in the "TMI-1 Restart Safety Analysis Report" it will be demonstrated through conservatism that safe operation of TMI-1 will be completely justified.

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V. Recommended Future Action

The "TMI-1 Restart Safety Analysis Report" will demonstrate that TMI-1 actual equipment actuation and performance are consistant with that used in the licensing basis analysis. The report will address potential effects of non-safety grade control systems in an adverse environment and assess conclusions reached in the original safety analysis. In addition, the report will assess the environmental effects of equipment required to maintain safe shutdown following accidents which cause an adverse environment. This effort can be closely coupled to the Abnormal Transient Operating Guidelines Program currently under consideration, and can focus upon additional operator training to recognize and respond to the impact of an adverse environment on non-safety grade control systems. The schedule for submittal of the Safety Assessment can be consistant with the current schedule for the Abnormal Transient Operating Guidelines Program (i.e., mid-1980).

A more detailed evaluation of potential effects of high energy line break accidents on non-safety control systems will be performed in the long term, with particular emphasis on the potential program areas identified in the safety assessment.

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TABLE I

TYPICAL EQUIPMENT RESPONSE DURING HIGH ENERGY LINE BREAKS

B&W 177 FA PLANTS

		Steam Line Break	Feedwater Line Break	Large LOCA	Small LOCA
I.	Reactor Power Control and Shutdown				
	Trip Function Utilized	High ϕ or Low RC Pressure	High RC Pressure	Reactor Trip Not Used	Low RC Pressure
	Time of Reactor Trip	1.1-8.0 sec.	8.2-13.4 sec.		
II.	Reactor Pressure Control				
	Time to PORV Actuation	PORV Not Actuated for Steam Line Break	4-8 sec.	PORV Response Not Important	PORV not assumed open
	Time at which PORV Closes		~20 sec.		
III.	Steam System Isolation and Pressure Control				
(1)	Steam Line Isolation Time	1.6-8.5 sec.	6.0-12.0 sec.	Code Safety Valves are Used in the Analyses for Conservatism	Code Safety Valves are Used in the Analyses for Conservatism
(2)	Time to Steam Relief Valve Opening	7.0-16.0 sec.	7.0-7.5 sec.		
(2)	Time for Steam Relief Valve Closure	20-30 sec.	25-30 sec.		
IV.	Feedwater System Isolation and Control				
(1)	Main Feedwater Isolation Time	19-34 sec.	~18 sec.	Analysis Conservatively Assumes a Loss of All Feedwater	Not Required
(1)	Auxiliary Feedwater Isolation Time	19-34 sec.	~18 sec.		Not Required
(2)	Auxiliary Feedwater Initiation Time	~40 sec.	~40 sec.		~40 sec.
(2)	Main or Auxiliary Feedwater Control	Maintain Minimum OTSG Level	Maintain Minimum OTSG Level		Maintain Preset OTSG Level

(1) Affected Steam Generator (2) Unaffected Steam Generator

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TABLE II

POTENTIAL ENVIRONMENTAL EFFECTS ON NON-SAFETY GRADE CONTROL SYSTEMS

	Licensing Basis Accidents					
Non-Safety Grade Control Systems	SLB Inside Containment	SLB Outside Containment	FWLB Inside Containment	FWLB Outside Containment	Large LOCA	Small LOCA
I. Reactor Power Control and Shutdown						
Control Rod Drive Control System	X	X	X	X	X	X
II. Reactor Pressure Control						
Power Operated Relief Valve	X	-	X	-	Y	X
Pressurizer Heaters	Y	Y	Y	Y	Y	Y
Pressurizer Spray	Y	-	Y	Y	Y	Y
III. Steam System Isolation and Pressure Control						
Turbine Trip/Turbine Stop Valves	-	X	-	X	-	-
Steam Line Isolation Valves*	-	X	-	X	-	-
Turbine Bypass/Atm Relief Valves**	X	X	X	X	X	X
IV. Feedwater System Isolation and Control						
Main Feedwater Control**	X	X	X	X	X	X
Main Feedwater Isolation*	X	X	X	X	Y	X
Auxiliary Feedwater Isolation	X	X	X	X	Y	X
Auxiliary Feedwater Initiation**	-	X	-	X	-	-
Auxiliary Feedwater Level Control**	X	X	X	X	X	X

* Affected Steam Generator
** Unaffected Steam Generator

\- Environmental Effects Cannot Occur Due to Location of Equipment (inside containment vs. outside containment)
Y Environment will not affect Safety Analysis Results
X Environment could affect Safety Analysis Results

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