Docket No. 50-271

REPORT ON FAILURE EFFECTS

OF EQUIPMENT NOT QUALIFIED FOR

ACCIDENT RELATED ENVIRONMENTS

Vermont Yankee Nuclear Power ... ation

Vernon, Vermont

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I. INTRODUCTION

This report presents the results of a survey conducted at Vermont Yankee to determine the impact on the analytical results presented in the Final Safety Analysis Report from failures of equipment subject to, but not qualified for, accident related environments.

This survey is being conducted in two phases as follows:

Phase I - This phase addresses the equivalent BWR systems identified in IE Information Notice 79-22. Phase I has been completed and the survey results as well as a description of the survey procedure is included in this report.

Phase II - This phase includes further studies to identify other germane systems, if any, that are not specifically identified in IE Information Notice 79-22. In addition, this phase addresses the impact of erroneous information displayed to plant operators that may result from failures to equipment not qualified for accident related environments. The objectives of this phase and results to date are included in this report.

II. LICENSE BACKGROUND

The NRC letter from the Director of Nuclear Reactor Regulation of September 17, 1979 requests within 20 days, informatin of affirmation which will enable the staff to determine, in light of the concerns discussed in that letter, whether or not Vermont Yankee's operating license should be modified, suspended or revoked.

Vermont Yankee's Final Safety Analysis Report was submitted in January 1970. It included all analysis considered applicable and required at that time.

III. PHASE I

(Systems identified in IE Information Notice 79-22)

A. PROCEDURE

Phase I of the Vermont Yankee survey was directed to systems performing functions equivalent to those identified in IE Information Notice 79-22. The influence of each applicable system on accident mitigation was considered. When a system/environment interaction was determined to have the potential for degradation of the accident analysis, further investigation took place. Such potential interactions are identified in Table No. 1. In those systems where potentials did exist, non-safety grade components exposed to accident environments were examined for their qualification to operate in that environment. Equipment not qualified was identified, and appears in the results of this report as "Areas of Concern".

Equipment whose qualifications could not be determined was considered not qualified. Equipment whose failure mode could not be determined was considered to fail in the mode most adverse to accident mitigation. If several items of nonqualified equipment were exposed to the same environment from any one accident, all are considered to fail in the most adverse mode.

Each "Area of Concern" was analyzed for its effects on accident mitigation. The impact of the concerns on the safety analysis appears in the evaluation of the scenario for each "Area of Concern".

TABLE 1

SYSTEMS OF EQUIVALENT FUNCTION TO IE NOTICE 79-22

Accident	Control Rod Drive System	Nuclear Boiler Pressure System	Feedwater Control System
L.O.C.A.	X	X	X
Main Steam Break Inside - Large	X	X	X
Small			
Reactor Building	X	X	X
Turbine Building			X
Feedline Break Inside	X	X	X
Reactor Building	X	X	X
Turbine Building			X

X = Area of possible impact to be investigated

Blank = No impact considered possible. Break in this system could not create an adverse environment for the control system.

III. B AREAS OF CONCERN

1. NUCLEAR BOILER PRESSURE RELIEF SYSTEM

System Description

In this boiling water reactor nuclear steam supply system, boiling takes place directly in the core. Steam from the reactor vessel is directed to the turbine. Pressure control is by the turbine throttle and bypass valves regulating backpressure. The system is designed to operate at a fixed pressure. Reactor power generation rate is controlled by recirculation rate, in accordance with power desired from the main generator, within the maneuvering band of the power operating range.

The pressure relief system consists of two spring loaded safety valves and four pilot operated safety and relief valves. The safety and relief valves are also operated by air pistons. By this means they are opened by remote automatic or manual, electric signal. This electropneumatic operation is used when the valves are automatically opened to lower reactor pressure during the Emergency Core Cooling Mode, so that low pressure coolant injection and/or core spray pumps can fill the vessel. This operation is described in VY FSAR Section 1.6.2.4 and 6.4.2.

The safety and relief valves are located in the primary containment, mounted on the steam lines. They discharge to the Suppression Pool.

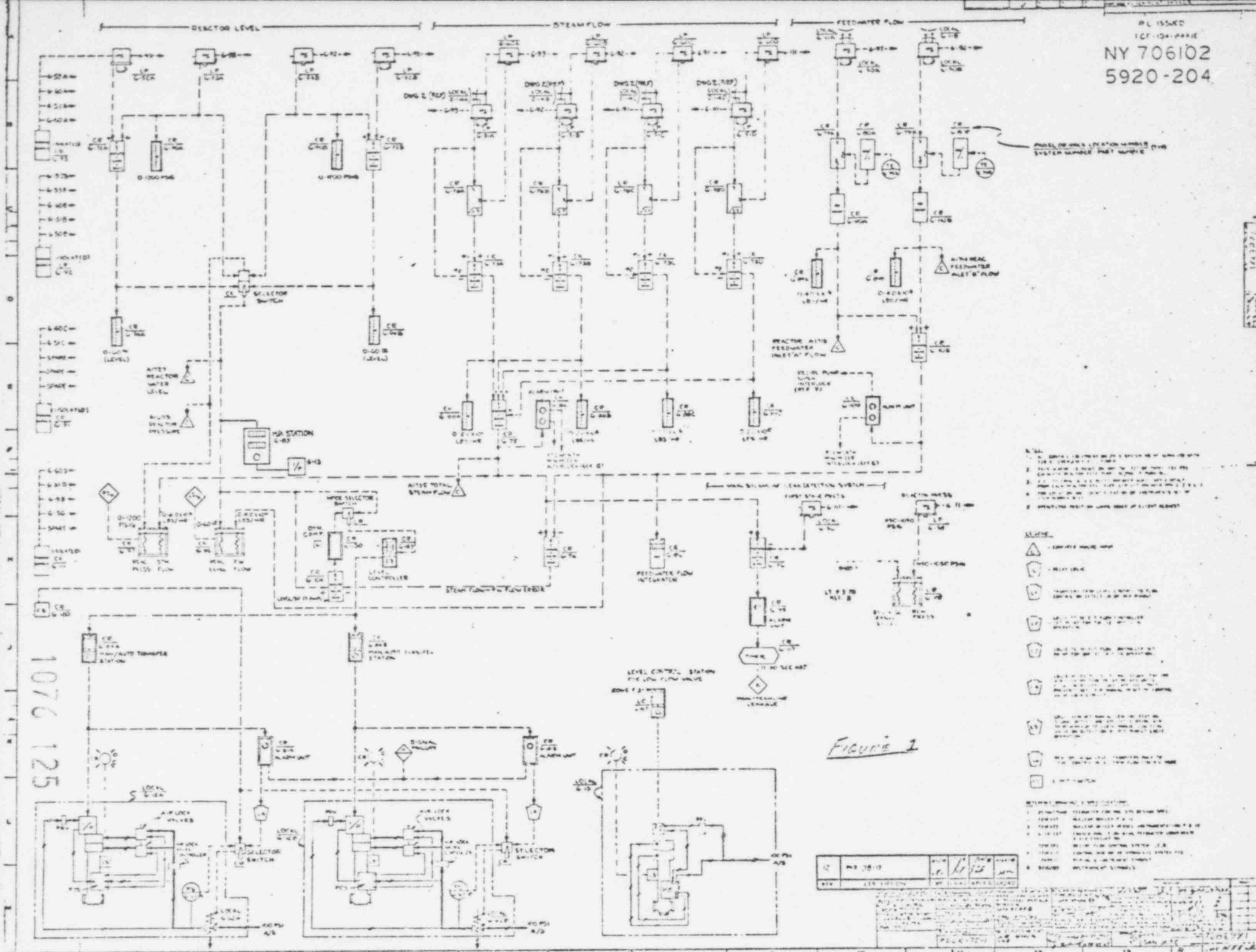
REACTOR LEVEL
STEAM FLOW
FEEDWATER FLOW
NY 706102
5920-204
SELECTOR SWITCH
H/A STATION
LEVEL CONTROLLER
FEEDWATER FLOW INTEGRATOR
LEVEL CONTROL STATION FOR LOW FLOW VALVE
SIGNAL FAILURE
AIR LOCK VALVES
ALARM UNIT
MAN/AUTO TRANSFER STATION
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Figure 1

The pressure switches that initiate remote automatic operation of the safety and relief valves are located on the 280 foot level of the reactor building. The relay logic and manual controls are located in the control room.

The equipment is qualified for the accident related environment in which it is located.

Concern

None

Safety Evaluation

The safety and relief valves and the control solenoids and cabling are located in the containment. They were designed to operate properly after exposure to the most severe environment that would exist as a result of a design basis loss-of-coolant accident. There is, therefore, no concern that these valves will operate inadvertently.

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2. FEEDWATER CONTROL SYSTEM

Description

The Vermont Yankee Feedwater Control System is a "three element control system". Input signals are: steam flow, feed flow, and reactor vessel level. These electric signals are combined to produce the control signal to the air operated feedwater regulating valves.

Reactor vessel level is sensed by differential pressure transmitters located on the 280 foot level in the reactor building, sensing vessel level and reference column level. Reference columns are inside the containment.

Steam flow is sensed by d/p cells located on the 250 foot level in the reactor building. The steam flow venturis are located inside the containment.

Feed flow is sensed by d/p cells located in the turbine building, measuring flow through the venturis in the steam tunnel.

The feedwater regulating valves are located in the turbine building in the feed pump room.

The feedwater electronic control circuitry is located in the control room. The reactor level and steam flow transmitters are qualified for the reactor building environment.

System interconnection is shown in Figure No. 1

Concern

There is concern that the environment resulting from a high energy pipe break might adversely affect the feedwater control system in such a way as to make the results of the accident more severe than that of the design basis analysis.

Safety Evaluation

A. Line Breaks Within the Primary Containment

Analysis of these breaks is bounded by the loss-of-coolant accident analysis in the VY FSAR. This analysis found that the worst possible accident is the double ended break of a recirculation line. Steam line or feed line breaks are less severe.

Only the level sensing columns and sensing lines are located within the containment. It is possible that the response of the level sensing could be affected by a high energy pipe break within the containment.

The loss-of-coolant accident analysis, when considering effects on the reactor core, assumed that feed flow went to zero in four seconds, since all AC power is assumed to be lost simultaneously.

When considering effects on the containment, the analysis assumed that feed flow went instantly to zero, which is the most severe case for the containment. The Mark I Containment Load Definition Report assumed that all the feedwater in the feed line above saturation temperature for final containment pressure flashed to steam and enters the reactor vessel.

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These analyses are believed to bound what would happen if an environmental effect caused the feedwater valves to close.

Based on the bounds of the above analysis, if AC power was not lost, and an environmental effect caused an increase in feedwater flow which continued, this would obviously serve to further mitigate the effects of the accident on the reactor core or containment.

For intermediate break and small break accidents, as analyzed in the Mark I Containment Load Definition Report, it was assumed that feedwater flow continued for ten minutes to maintain level. If an increase in flow occurs, this is assumed to be no worse a consequence because it will only result in a higher vessel level. If a flow decrease occurs, this is assumed to be conservative, since if the break uncovers, the flow rate out of the vessel should decrease.

It is concluded that even if the Feedwater Control Valve position is influenced by an environmentally induced effect within the containment, the consequences will be no more severe, or less severe than those assumed in analyses already done.

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B. Line Breaks Outside the Containment

1. Steam Line Break

The Safety Analysis considered a complete double-ended steam line break where the feed flow went to zero in four seconds. The core was protected. Environmental effects on the feedwater control system causing a decrease in flow are a less severe feed flow situation (slower decrease) than the safety analysis considered. Obviously, if feed flow does not decrease this rapidly, coolant inventory in the vessel will be greater. Due to swell effects, the basic analysis considered two phase flow in the steam lines. The mass loss rate in the analysis far exceeds the rate at which feed pumps could possibly add water to the vessel. These last facts mean that any possible feed flow increase due to an environmental effect will not have an adverse effect on the accident.

2. Feed Line Break

The Safety Analysis considered a complete loss of all feedwater pumps as the most severe loss of feed flow accident, and an instantaneous step increase in feed flow to maximum possible demand as the most severe increase of feed accident. In both cases no reactor core limits were exceeded. Thus, any feed flow increase or decrease caused by an environmental effect on the feedwater control system will not be a more severe condition than that already considered.

3. CONTROL ROD DRIVE SYSTEM

System Description

The control rod drive mechanisms are hydraulically operated by water taken from the condensate system, which is the same water that is fed to the reactor. The mechanism withdraws the rod by pulling it down and inserts it by pushing it up.

Because the rod is withdrawn down, the mechanism is designed with Collet Fingers to support and lock the mechanism in place. The Collet Finger tips are inserted into notches in the index tube attached to the rod. They carry the full weight of the rod when it is not moving, and are designed so that they cannot be withdrawn from the notches unless the weight of the control rod is lifted off the fingers. The Collet Fingers may then be withdrawn and held withdrawn by water pressure. See Figure No. 2.

There is an insert water line and a withdraw water line to each mechanism. One is pressurized, the other vented to the return path, depending on the direction of motion. Normal flow to the mechanism is controlled by four solenoid operated valves, a pressurizing and venting valve for each line. For SCRAM, air operated valves supply and vent water, bypassing the solenoids. The water comes from an accumulator system in this case.

For normal control rod motion, the solenoids are controlled by the rod control circuits. For a rod withdrawal, the insert solenoids are first energized to insert the rod and lift it off the Collet Fingers. The withdraw solenoids are then energized and the insert solenoids de-energized. Withdraw water holds the fingers retracted while the rod is withdrawn.

The mechanism is on the bottom of the reactor vessel. The solenoids and SCRAM valves are on the hydraulic control unit for each rod, located on the ground floor of the reactor building, near the containment, on the north and south sides. The control circuits and equipment are located in a panel in the main control room.

The control rod drive system does not have any automatic rod withdrawal function. The rods can only be withdrawn manually, one at a time. The manual control devices are located in the control room and cannot be subject to environments caused by high energy line breaks.

Concern

There is concern that failures in the control system induced by a high energy line break might cause a failure in the control system that would result in an unplanned rod withdrawal.

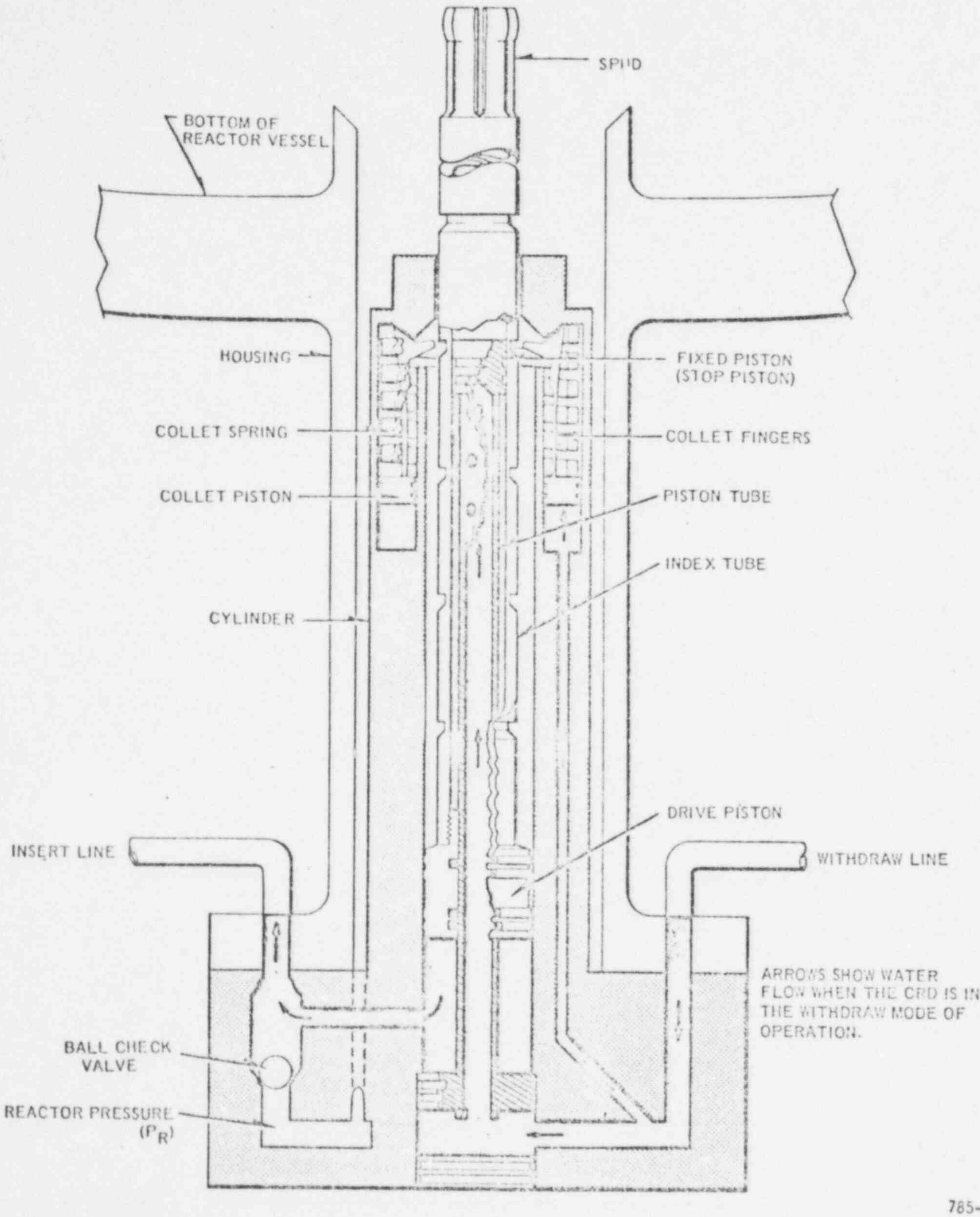

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Figure 2-1. CRD (SCHEMATIC DIAGRAM)

Figure No 2

Safety Evaluation

Since there is no automatic rod withdrawal capability associated with this system, and because all controls associated with rod withdrawal are remote from high energy line breaks, there can be no environmentally induced failure which results in rod withdrawal.

The only high energy lines able to impact on the Control Rod Drive System are those in the steam tunnel. Any steam released there will not directly impact the hydraulic control units, for one half the control rods, that are located on the south side of the reactor building, because of the torturous path through the CRD repair room that the steam must follow. The HCU's are located approximately 40 feet from the CRD repair room door, on the open first floor of the reactor building.

The steam tunnel has blow-out panels that release steam to the turbine building, and a blow-out panel in the door to the reactor building that releases steam there.

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The two SCRAM valves for each control rod are metal bodied, spring opened, air to close valves. The SCRAM pilot valves controlling the air to the SCRAM valves are solenoid valves, energized to supply air. When they de-energize and change position, they vent air from the SCRAM valves, opening them.

The location of the Hydraulic Control Units makes it highly unlikely that they would experience a very detrimental environment (i.e. high temperature and humidity) very soon after a high energy line break in the steam tunnel. Thus, rods would have scrammed long before local environmental conditions possibly bacame severe.

The design of the Hydraulic Control Units SCRAM and SCRAM pilot valves is such that if they fail a SCRAM will be caused.

It is Vermont Yankee's judgement that a high energy line break in the steam tunnel will not have an adverse impact on any equipment such that it will make the effects of this accident more severe.

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C. SUMMARY OF RESULTS

The areas of concern where potential interactions between non-safety related systems and adverse environments exist have been investigated. It was concluded that some non-safety system responses could possibly result. Consideration of the possible responses and their consequences has led to the conclusion, in Vermont Yankee's judgement, that the ability of plant safety systems to safely mitigate the effects of the high energy pipe breaks considered will not be decreased. It is Vermont Yankee's judgement that the conclusions of the Final Safety Analysis are valid.

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IV. PHASE II

A. OBJECTIVE

It will be the objectives of this Phase of the study to:

1. Survey other non-safety grade systems in the plant to determine whether or not failures resulting from accident related environments could have an adverse effect on safety system performance. If adverse effects are found possible, corrective action will be initiated. This work will be guided by the genericc matrix prepared by General Electric Company and supplied to the Owners Group. (Table 2)

2. Determine the impact of erroneous indication displayed to operators which could result from environmentally induced failures

3. Determine what corrective action for items found in "2" is required.

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B. RESULTS TO DATE

The Phase II study has already begun. To date, it has been determined that the reactor head vent valves, identified as a potential problem in the GE generic study, are also potentially an area of concern for Vermont Yankee. In the Vermont Yankee plant there are two valves in series in the head vent line. They are air operated to open, spring closed. The solenoids which control the air are not qualified for the post LOCA containment environment. Even if these valves were to both open, the GE analysis shows that PCT in a LOCA will be increased only 10^{o}F.

The solenoid operated air valves that control the air to the head vent valves (V-2-17,18) are energized to open. Thus an environmentally induced failure would have to energize both solenoids at the same time. Since only the solenoids and cables are in the containment, with control switches and contacts in the control room, such an environmental failure is not considered a credible event.

Vermont Yankee considers that the design of the reactor held vents V-2-17 and V-2-18 is satisfactory and safe as it is.

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TABLE 2 ENVIRONMENTAL INTERACTION

		Main Steam Line				FeedWater			LOCA inside drywell		RWCU	Isolation Condenser	RCIC	HPCI
	Location	Inside Small	Inside Large	Reactor Bldg	Turbine Bldg	Inside	Reactor Bldg	Turbine Bldg	SML	LRG	Outside	Outside	Outside	Outside
Recirc System.														
• Pumps	DW	2	2	4	4	2	4	4	2	2	4	4	4	4
• Valves & Operators	DW	2~3	2~3			2~3			1~3	1~3		4		
• MG Sets	TB/RB	4	4			4			4	4		2~4		
• MCC	TB/RB											2~4		
• Flow Control Syst.	CR											4		
• Control Inst. Transmitters	RB											2~4		
Feedwater Delivery System.														
• Flow Elements	TB	4	4	4	2	4	4	2	4	4	4	4	4	4
• Level	DW/RB	2	2		4	2		4	2	2				
• Pumps	TB	4	4		2	4		2	4	4				
• Valves & Operators	TB				2			2						
• MCC	TB				4			4						
• Flow Control System	CR				4			4						
• FW Heating	TB				2			2						
• Instrument Air	TB				2			2						
• Control Inst. Transmitter	RB/TB			2	2		2	2						
Turbine Pressure Control.														
• ByPass Valves	TB	4	4	4	2	4	4	2	4	4	4	4	4	4
• Pressure Sensors	TB				2			2						
• Control System	CR				4			4						
Neutron Monitoring System.														
• LPRMs & Cables	DW/RB	2	2	2	4	2	2	4	2	2	2	4	4	4
• APRMs & Cables	DW/RB	2	2	2	4	2	2	4	2	2	2	4	4	4
• RPIS/Rod Block Monitor	DW/RB	2	2	2	4	2	2	4	2	2	2	4	4	4
• TIP	DW/RB	2	2	2	4	2	2	4	2	–	2	4	4	4
Reactor Protection System.														
• Turbine Scram	TB	4	4	4	2	4	4	2	4	4	4	4	4	4
• MG Set	TB	4	4	4	4	4	4	4	4	4	4	4	4	4
Reactor Manual Control System.	RB/CR	4	4	4	4	4	4	4	4	4	4	4	4	4
SRV System (non ADS)	DW/RB	3	3	3	4	3	3	4	3	3	4	4	4	4
RBCCW System	RB	4	4	2	4	4	2	4	4	4	2	2	4	4
RWCU	DW/RB	3	3	2	4	3	2	4	3	3	2	2	2	2

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TABLE 2 ENVIRONMENTAL INTERACTION (cont'd)

	Location	Main Steam Line: Inside Small	Main Steam Line: Inside Large	Main Steam Line: Reactor Bldg	Main Steam Line: Turbine Bldg	Feed Water: Inside	Feed Water: Reactor Bldg	Feed Water: Turbine Bldg	LOCA: Sml	LOCA: Lrg	RWCU: Outside	Isolation Condenser: Outside	RCIC: Outside	HPCI: Outside
Suppression Pool • Temperature Monitoring • Level Monitoring	RC/DWBS & B/DWBS	2 or 4 / 3 4	2 or 4 / 3 4	2 or 4 / 3 2	4 / 4 4	2 or 4 / 3 4	2 or 4 / 3 2	4 / 4	2 or 4 / 3	2 or 4 / 3	4 / 4	4 / 4	4 / ~~4~~	4 / ~~4~~
Circulating Water System (Non Safety)	INTAKE/TB	4	4	4	2	4	4	2	4	4	4	4	4	4
HVAC System	All	2	2	2	2	2	2	2	2	2	2	2	2	2
Non IE Battery System	TB	4	4	4	4	4	4	4	4	4	4	4	4	4
AC Auxiliary Electric	RB/TB	4	4	4	4	4	4	4	4	4	4	4	4	4
Condensate Transfer & Storage	TB	4	4	4	~~2~~	4	4	2	4	4	4	4	4	4
Main Turbine & Controls	TB	4	4	4	2	4	4	2	4	4	4	4	4	4
Main Condenser & Control	TB	4	4	4	2	4	4	2	4	4	4	4	4	4
Instrument Air System • Compressors • Piping & Controls	TB / TB/RB/DW	4 / 2	4 / 2	4 / 2	2 or 4 / 2	4 / 2	4 / 2	2 or 4 / 2	4 / 2	4 / 2	4 / 2	4 / 2	4 / 2	4 / 2
Fire Protection System	TB/RB/CR	4	4	1,2 or 4	1,2 or 4	4	1,2 or 4	1,2 or 4	4	4	1,2 or 4	1,2 or 4	1,2 or 4	1,2 or 4
CRD Hydraulic System (Non Scram)	RB	4	4	2	4	4	2	4	4	4	4	4	4	4
RV Head Vent	DW	2	2	4	4	2	4	4	1	1	4	4	4	4
SLC System	DW/RB	3 2	3 2	3 2	4	3	2	4	3	3	3	2	3	2
Remote Shutdown														
Air Coolers														
Sump Pump														

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TABLE 2 (CONT'D)

SYSTEM	ANY HIGH ENERGY BREAK
LIGHTING	5
COMMUNICATIONS	
SERVICE AIR	
EQUIP DRAIN PIPING	
DRYWELL TEMP. MONITORING	
UNDER VESSEL MAINTENANCE EQUIP.	
PROCESS COMPUTER	
AREA RADIATION MONITORING	
PROCESS RADIATION MONITORING (NON SAFETY PART)	
SAMPLING SYSTEMS	
MAINTENANCE MONORAILS	
ENVIRONS MONITORING	
DEMINERALIZED WATER	
POTABLE WATER	
SCREEN WASH	
HYDROGEN COOLING	
CONDENSER PRIMING	
TBCCW	
STATOR COOLING	
OFFGAS	
RADWASTE	

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TABLE 2.1 (CONT'D)

1 = ENVIRONMENTAL INDUCED MALFUNCTION MAY PROVIDE ADVERSE RESPONSE, i.e. INCREASE IN PREVIOUSLY REPORTED PEAK

- DRYWELL PRESSURE
- WETWELL PRESSURE
- SUPPRESSION POOL TEMPERATURE
- FUEL CLAD TEMPERATURE

2 = ENVIRONMENTAL INDUCED MALFUNCTION WILL NOT PROVIDE ADVERSE RESPONSE

3 = SYSTEM IS QUALIFIED FOR ADVERSE ENVIRONMENT

4 = SYSTEM WILL NOT EXPERIENCE ADVERSE ENVIRONMENT

5 = NO CONCEIVABLE SYSTEM FAILURE AN AFFECT RESPONSE

NOTE: CATEGORY "1" AND "2" ENTRIES, WHICH REFLECT THE EFFECT ON ECCS AND CONTAINMENT ANALYSES GIVEN THE FAILURE OF THE SUBJECT SYSTEM OR COMPONENT HAVE BEEN VERIFIED TO BE GENERICALLY TRUE BY GENERAL ELECTRIC.

CATEGORY "3", "4" AND "OR" ENTRIES WHICH ADDRESS WHY A GIVEN NON-SAFETY SYSTEM WOULD NOT FAIL, ARE OBVIOUSLY PLANT UNIQUE AND MUST BE VERIFIED INDEPENDENTLY BY EACH UTILITY.

CATEGORY "5" ITEMS, WHICH SEEM EVIDENT, MUST NEVERTHELESS BE VERIFIED INDEPENDENTLY BY EACH UTILITY.

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