CENTRAL FILES

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

June 22, 1979

Mr. Boyce H. Grier, Director
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: IE Bulletins 79-06A and 79-06A Revision I entitled "Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident" - additional information requested during staff review of responses
R. E. Ginna Nuclear Power Plant, Unit No. 1
Docket No. 50-244

Dear Mr. Grier:

Attached is the reply of the Rochester Gas and Electric Corporation to a request from members of the NRC Staff for additional information as a result of the Staff review of our letter dated April 28, 1979 which responded to IE Bulletins 79-06A and 79-06A Revision 1.

Our responses have been prepared by a continuing Task Force at Ginna Station. The questions we have received from the NRC Staff and our responses are numbered to correspond to the bulletins' action items.

Very truly yours,

L. D. White, Jr.

L. D. White, Jr.

A002
C/P

Att.

xc: NRC Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

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Item 2. PROVIDE DATE FOR COMPLETION OF YOUR REVIEW. UPON COMPLETION OF YOUR EFFORTS WE REQUIRE THAT YOU SUBMIT A SUMMARY OF THE RESULTS INCLUDING REVISIONS MADE TO OPERATING PROCEDURES. IN ADDITION, SUMMARIZE THE REVIEW RESULTS AND ACTIONS TAKEN WITH REGARD TO THE NATURAL CIRCULATION MODE OF OPERATION.

Substantial progress has been made in completing the procedure changes described in our April 28, 1979 letter. In addition several new procedures have been developed.

RCS accident emergency procedures have been reviewed with particular attention being paid to the possibility of void formation. Actions to prevent inadvertant formation of voids, as described in our April 28 letter, have been incorporated in the procedures. Specific emergency procedures which have been modified include: E-1.1, Safety Injection Initiation; E-1.2, Loss of Coolant Accident; E-1.3, Steam Line Break Accident; E-1.4, Steam Generator Tube Rupture Accident.

A new emergency procedure, E-1.6, CVCS Breaks, is being developed. This procedure will consolidate three existing procedures and will, we believe, provide for better operator response to breaks in the CVC system. We expect this procedure to be completed and approved for use by July 2,1979.

Two procedures dealing with heat removal from the primary system and potential void formation are being prepared. Procedure O-8, Natural Circulation, will provide instructions on identifying whether or not heat is being removed from the primary system by Natural Circulation. Key parameters are identified and instructions for establishing, or reestablishing, natural circulation are provided. Information from the pre-operational tests which were performed at Ginna is employed as well as guidelines from Westinghouse. We expect that this procedure will be completed and approved for use by July 2, 1979. In the interim, operator training sessions have covered, and will continue to address, key features of natural circulation. (see information from lesson plan, Attachments 2, 3 & 4).

A new procedure, E-1.5 Void Formation in the RCS, has been written and is currently undergoing plant review prior to approval. Final approval is expected by June 25, 1979. This procedure will provide guidance on how to eliminate voids from the RCS. In addition, guidance on removing heat from the RCS if neither forced nor natural circulation is available and if the break flow in a LOCA is not sufficient to remove decay heat.

Finally, operator training sessions will continue to include presentations on the most recent information (see lesson plan outline Attachment 2) which is available from Westinghouse and elsewhere concerning void formation.

Item 4. PROVIDE SUFFICIENT INFORMATION AS TO WHICH LINES ARE OR ARE NOT ISOLATED SO THAT A CONCLUSION CAN BE REACHED AS TO WHETHER ALL LINES - EXCEPT THOSE NEEDED FOR NEEDED SAFETY FEATURES OR COOLING CAPABILITY - ARE ISOLATED. INCLUDE A DISCUSSION OF THE OPERABILITY OF REACTOR COOLANT PUMPS UNDER THIS ISOLATED CONDITION.

Table 1 lists all valves which are isolated (closed) on either a containment isolation signal or a containment ventilation isolation signal. A containment isolation signal closes the valves shown as items 1 through 28 in Table 1, trips the containment sump pumps, and initiates a containment ventilation isolation signal. Containment ventilation isolation closes the valves shown as items 29 through 38 in Table 1 and trips the purge supply and exhaust fans.

Remaining lines, which are not isolated on a containment isolation or containment ventilation isolation signal, are those required for safety functions. In addition, component cooling water to the reactor coolant pumps and seal injection to the reactor coolant pumps are not isolated. Therefore, the reactor coolant pumps remain operable under this isolated condition.

In our response of April 28, we stated that safeguard logic schemes were being reviewed in order to identify whether any changes were required with regard to containment isolation. That review has been completed with the conclusion that no changes were necessary.

We have further verified that all lines which penetrate containment that are not required for safety features or reactor coolant pumps are isolated by either locked valves, normally closed valves or automatic valves that close on an isolation signal.

Item 7 a. PROVIDE A SCHEDULE FOR COMPLETION OF THE REVIEW OF OPERATING PROCEDURES AND TRAINING INSTRUCTIONS, INCORPORATING SUCH MODIFICATIONS AS ARE NECESSARY TO ENSURE THAT OPERATORS WILL NOT OVERRIDE AUTOMATIC ACTIONS OF ENGINEERED SAFETY FEATURES, UNLESS CONTINUED OPERATION OF ENGINEERED SAFETY FEATURES WILL RESULT IN UNSAFE PLANT CONDITIONS IN ORDER TO COMPLY WITH ITEM 7.A OF THE BULLETIN. CLARIFY THE MEANING OF WHAT IS MEANT BY REQUIRING CONCURRENCE BY TWO LICENSED INDIVIDUALS TO OVERRIDE EMERGENCY SAFETY FEATURES.

The following Emergency procedures have been reviewed, modified and issued as necessary to ensure proper operator actions.

a). E-1.1 (Safety Injection System Actuation)
b). E-1.2 (Loss of Coolant Accident)
c). E-1.3 (Steam Line Break Accident)
d). E-1.4 (Steam Generator Tube Rupture)

Specific guidelines have been included which closely reflect the Westinghouse recommendations. The revised emergency procedures are currently being presented to the operators by the Training Department. This presentation will be completed by July 13, 1979. Also, changes to procedures are reviewed by licensed personnel via the procedure acknowledge book.

In addition, Administrative Procedure, A-54.1 (Licensed Personnel Authority) was modified by Ginna Station procedure change notice No.79-1168 to state that two licensed operators shall agree on any overriding before the overriding action is executed on any safeguard system active component. This administrative action is designed to meet the guidelines of IE Bulletin 79-06A in allowing the operator to override any component in the safeguards system if the continued operation of that component will result in unsafe plant conditions while ensuring that undesired overrides, such as stoppage of SI flow to the RCS when it is required for core cooling, is precluded.

Item 7 b. YOUR RESPONSE TO ITEM 7.B APPEARS TO BE INADEQUATE WITH REGARD TO THE REQUIREMENTS OF ITEM 7.B OF THE BULLETIN. PROVIDE ASSURANCE THAT OPERATING PROCEDURES WILL BE MODIFIED TO KEEP HIGH PRESSURE INJECTION AND CHARGING PUMPS IN OPERATION IN ACCORDANCE WITH THE CRITERIA SPECIFIED IN ITEM 7.B OF THE BULLETIN. PROVIDE A SCHEDULE FOR COMPLETION OF THE REVIEW OF OPERATING PROCEDURES INCORPORATING SUCH MODIFICATIONS AS ARE NECESSARY TO COMPLY WITH ITEM 7.B OF THE BULLETIN.

The guidelines set forth by the NRC in the IE Bulletin 79-06A Rev. 1 for stopping SI components are not consistent with the guidelines that we have received from our NSS supplier, Westinghouse. It is RG&E position at present to closely reflect the Westinghouse recommendations. As stated in our response to item 2, procedures E-1.1 through E-1.4 have already been revised.

In addition more information will be forthcoming from Westinghouse, as a result of their response to a letter from D.F. Ross, NRC to T.M. Anderson, Westinghouse, dated June 4, 1979.

Item 14 of that letter requests, in part:
Provide the results of an analysis of the effects of different HPI termination criteria on the course of small LOCA's. Specifically for each small break LOCA analyzed, compare the effects of the NRC HPI termination criteria (as stated in IE Bulletins 79-06A and 79-06A, Rev. 1 item 7 (b)), to those for the HPI termination criteria which have been recommended to licensees with Westinghouse designed operating plants.

Upon receipt of additional or revised guidelines from Westinghouse which are applicable to our plant, with its lower head high head safety injection pumps, we will promptly revise the applicable procedures.

Item 7 c. YOUR CRITERIA FOR TRIPPING REACTOR COOLANT PUMPS IS INCONSISTENT WITH THE PROVISIONS OF ITEM 7.C OF THE BULLETIN. PROVIDE ASSURANCE THAT OPERATING PROCEDURES WILL BE MODIFIED TO KEEP REACTOR COOLANT PUMPS IN OPERATION IN ACCORDANCE WITH ITEM 7.C OF IE BULLETIN. PROVIDE A SCHEDULE FOR COMPLETION OF THE REVIEW OF OPERATING PROCEDURES INCORPORATING SUCH MODIFICATIONS AS ARE NECESSARY TO COMPLY WITH ITEM 7.C OF THE BULLETIN.

The NRC requirements for RCP operation in an accident condition are not consistent with the Westinghouse recommendation. At present RG&E is using the Westinghouse recommendation for stopping RCP's in an accident situation. Westinghouse's guideline is to stop all running RCP's when system pressure is $\leq$ 1550 psi and the SI system is delivering water to the RCS. This recommendation has been incorporated in the immediate actions of the Loss of Coolant, Safety Injection System Actuation and S/G Tube Rupture procedures (E-1.1, 1.2, and 1.4). However, 1500 psi has been used because the wide range pressure indicator on the main control board has a major division at 1500 psig. Therefore, this pressure is chosen as an aid to the operator.

In the Steam Line Break Accident procedure the RCP's are tripped immediately once the accident is identified to slow down the heat transfer rate to the steam generator. This condition is being discussed on an continuing basis with Westinghouse. It should be noted that analysis of main steam line breaks show a rapid RCS pressure decrease with the pressure dropping below 1500 psia withing 25 sec for large breaks and within about 2 minutes for a break equivalent to a stuck open relief valve. Thus, we believe the criteria in this procedure are consistent with the Westinghouse guidelines.

Item 7 d. PLEASE VERIFY THAT THE IDENTIFIED PARAMETERS HAVE BEEN INCLUDED IN APPROPRIATE OPERATING PROCEDURES.

Pressurizer level has been eliminated in the diagnostic scheme for the identification of accident category. Pressurizer level has also been removed as a symptom from all RCS accident procedures except CVCS break. The operator is directed to other indications for accident identification. The diagnostic tree is attached.

The indications listed in the initial RG&E response to NRC bulletin 79-06A Rev. 1 were incorporated from the Westinghouse response to the bulletin.

Wide range RCS temperature & pressure
Steam pressure
Steam generator water level
Containment pressure
RWST level
Condensate storage tank level
Pressurizer water level
Boric acid storage tank level

All the parameters listed are not indicative of RCS water inventory. Each of these parameters except the condensate storage tank level are however, procedurally addressed in the LOCA and steam line break procedure.

The Boric acid storage tank level is addressed in LOCA and steam line break procedures in the SI pump suction change over verification to the RWST. The remaining indications listed above are symptoms of the accident, are used during the accident as operator aids for verification of safe plant conditions, are required for operator action during the accident and/or are alarmed on the control board. (The attached diagnostic tree provides an example of how the operator is instructed to consider a variety of plant indications in his evaluations).

AFTER REACTOR TRIP & "S" SIGNAL

Low or falling pressurizer pressure

Abnormally low steam pressure in one or both steam generators indicates steam break

Verify by checking for:

1). Indication of a uncontrolled cooldown

2). Steam line isolation on the effected loop

3). S.I. on low steam pressure

4). A possible first out annunciation of
 a). steam/feedwater flow mismatch or
 b). low-low steam generator water level

Go to Emergency procedure E-1.3 Steam Line Break

Rising or normal steam pressure in both steam generators indicates loss of coolant or tube rupture

Either increasing containment pressure or containment high radiation alarm or rising sump water level indicates a loss of coolant

Go to Emergency procedure E-1.2 Loss of Reactor Coolant

Air ejector radiation alarm or steam generator blowdown radiation alarm or possibly observed differential rate of rise of steam generator levels

Go to Emergency procedure E-1.4 Steam Generator Tube Rupture.

Item 8. PLEASE PROVIDE YOUR SCHEDULE FOR COMPLETING REVIEWS OF ALIGNMENT REQUIREMENTS AND PROCEDURES CONTROLLING MANIPULATION OF SAFETY-RELATED VALVES. SUBMIT A SUMMARY OF THE RESULTS OF THE REVIEWS AND ANY REVISIONS NECESSARY WITHIN TWO WEEKS AFTER COMPLETION OF THE REVIEWS.

ALSO REVIEW PLANT PROCEDURES AND REVISE THEM AS NECESSARY TO ENSURE THAT LOCKED SAFETY-RELATED VALVES ARE SUBJECTED TO PERIODIC SURVEILLANCE. SUBMIT A SUMMARY OF THE RESULTS OF THE REVIEW.

All Periodic Test (PT) procedures that concern the Engineered Safety Features Systems have been reviewed and updated where applicable, to ensure proper valve realignment following testing. Each procedure directs test personnel to properly realign systems. Further, additiional steps have been added to assure the systems have been realigned for operation. This additional verification of realignment of systems is performed by personnel other than Testing Personnel, usually the Operation Department. The Periodic Test (PT) procedures are performed on a scheduled basis according to Technical Specification requirements, and prior to and following Maintenance of Engineered Safety Features Systems. The Periodic Test (PT) procdure changes were reviewed and approved by the plant operating review committee (PORC). Table 2 lists the PT procedures which have been revised to incorporate the additional steps and describes the changes.

New procedures have been developed and incorporated in Plant Operations to assure proper system valve line-up. These system valve position verification procedures are performed on a regularly scheduled basis. The performance of these procedures is in addition to the valve verification steps included in the Periodic Test procedures.

There are no Technical Specification requirements regarding locked valve surveillance. An Administrative procedure, (A-52.2), governs the control of all locked valves at our facility, and states: "The purpose of this procedure is to describe the requirements for a locked valve, authorities involved, documentation required and instructions relative to locked valve operation." The purpose of locked valves at our facility is to provide control of equipment and to maintain reactor safety, engineered safety features system alignment and personnel safety. The shift foreman has the authority to issue a key to unlock a valve and the head control operator has the duty to maintain valve status, if changed, in the locked valve operations log.

Surveillance of these locked valves is accomplished by valve verification steps in Periodic Tests (PT's) following System Testing and by regularly scheduled system valve verification procedures noted below.

S-30.1	Safety Injection System	valve	position	verification.	
S-30.2	RHR	"	"	"	"
S-30.3	Containment Spray	"	"	"	"
S-30.4	Auxiliary Feedwater	"	"	"	"
S-30.5	Standby Aux. FW	"	"	"	"
S-30.6	Safeguard valve position verification (inside CV)				

Item 9. PLEASE PROVIDE A COMPLETE RESPONSE TO ITEM 9 AND IDENTIFY THOSE ISOLATION VALVES WHICH MAY BE REPOSITIONED AS A RESULT OF RESETTING CONTAINMENT ISOLATION.

Table 1 lists all lines which are designed to transfer potentially radioactive fluids from containment. The containment isolation signal closes valves 1-28 (Table 1), trips the containment sump pumps, and initiates a containment ventilation isolation signal. Containment ventilation isolation closes valves 29 thru 38 (Table 1) and trips the purge supply and exhaust fans.

It should be noted that the resetting of the SI signal does not reset containment isolation or containment ventilation isolation and therefore does not reposition any valves.

Valves (by number from Table 1) 3, 15, 16, 17, 19, 20, 21, 22, 23, 24, and 26 are the valves that are normally open and would return to the open position upon reset of Containment Isolation unless additional actions were taken to preclude this. Note that number 19, 20, 21, and 22 will not reopen if a high radiation condition in the steam generator blowdown system exists. Present LOCA and Steam Line Break procedures require that the operators place all containment isolation valve switches in the "closed" position, prior to resetting containment isolation.

Furthermore, emergency procedures have been modified to instruct the operator to place the containment sump "A" pumps in the pull-stop position prior to resetting the containment isolation signal.

In the S/G tube rupture accident, it is our position that normal charging, letdown, spray and power operated relief valve operations need to be established quickly and that no radioactive transfer is possible from containment to the environs, except through the secondary steam lines. Therefore, containment isolation is reset after the accident is identified as a S/G tube rupture and the containment vessel sump pumps are pull stopped, and the letdown isolation and reactor coolant pump seal return containment isolation valves are closed.

Prior to the TMI Incident, there were several letters exchanged between RG&E and the NRC which dealt with safety actuation circuits and their overrides. These letters are provided in Attachment 1 to this letter for your convenience. Furthermore, containment ventilation isolation does not automatically reset when SI or containment isolation is reset. The reset of containment ventilation isolation requires the use of a key which is held by the Shift Foreman.

Item 10 a. PLEASE PROVIDE THE BASIS FOR THE CONCLUSION THAT EXISTING PROCEDURES FOR VERIFICATION OF OPERABILITY OF SAFETY-RELATED SYSTEMS ARE SUFFICIENT.

From the basis of the Technical Specifications Sections 3.3 and 4.5, the active components (pumps and valves) of safety related systems are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required) and would result in increased wear of long periods of time.

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures.

To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component. This is applicable to the safety injection pumps, residual heat removal pumps, and their associated valves (T.S. 3.3.1.2), and to the containment spray pumps and valves (T.S. 3.3.2.2). The two containment spray pumps also must be tested to demonstrate operability before initiating maintenance on an inoperable charcoal filter unit (T.S. 3.3.2.2). With one diesel generator out of service, the remaining diesel generator is run continuously provided such operation is not in excess of seven (7) days (total for both diesels) during any month, and provided the station transformer is in service (T.S. 3.7.2).

Testing of redundant components on the auxiliary feedwater system is not required by our Technical Specifications. However, Maintenance and Turbine Plant Operations procedures have been revised to notify the Results and Test Department to test the redundant component(s) prior to initiating repair of the inoperable component. Table 2 lists the procedures which have been revised to include this requirement.

Item 10 b. PLEASE PROVIDE YOUR SCHEDULE FOR COMPLETING THE REVIEW AND MODIFICATION OF PROCEDURES. WITHIN TWO WEEKS AFTER COMPLETING THIS EFFORT, PLEASE SUBMIT A SUMMARY OF THE RESULTS OF THE REVIEW AND THE ACTIONS TAKEN.

Maintenance procedures have been reviewed and are sufficient to ensure the operability of safety related systems after they are returned to service following maintenance. After maintenance, the Results and Test Department is notified and performs the applicable periodic tests to assure that the system is operable.

Periodic test procedures have been reviewed and changed as noted in Table 2. After testing of a safety related system, an additional verification by non-test personnel is required. All safety related valves manipulated during the test are checked to ensure they are in their proper position. After this verification, the completed test procedure must be reviewed by the Head Control Operator and approved by the Shift Foreman.

Item 10 c. PLEASE IDENTIFY THE LEVEL OF AUTHORITY REQUIRED FOR REMOVING AND RETURNING SYSTEMS TO SERVICE AND DESCRIBE THE METHOD USED FOR TRANSFERRING INFORMATION ABOUT THE STATUS OF SAFETY-RELATED SYSTEMS AT SHIFT CHANGE.

The Shift Foreman must approve the removal from and the return to service of all safety related systems as required by Administrative procedure A-52.4, " Control of Limiting Conditions for Operating Equipment". A procedural change, approved on April 30, 1979, to A-52.4 also requires notification of the Head Control Operator.

The transfer of information about the status of safety related systems at shift change is accomplished through Administrative procedure A-52.1, "Shift Organization Relief and Turnover", which lists the requirements for shift turnover.

Item 12. PLEASE PROVIDE A SCHEDULE FOR WHEN PROCEDURES DEALING WITH HYDROGEN GAS IN THE PRIMARY COOLANT SYSTEM WILL BE PREPARED.

A procedure to deal with hydrogen gas in the primary coolant system is being prepared and will be approved by PORC for use by July 16, 1979.

TABLE 1

CONTAINMENT ISOLATION VALVE CLOSURE

1.	MOV-313	Seal water return isolation valve
2.	MOV-813 & 814	Supply & return component cooling water to reactor support cooling
3.	AOV-371	Letdown isolation valve
4.	AOV-539	Pressurizer relief tank to Gas Analyzer
5.	AOV-846	Master N_2 stop to Accumulator
6.	AOV-951	Pressurizer steam space sample
7.	AOV-953	Pressurizer Liquid space sample
8.	AOV-955	"B" Loop hot leg sample
9.	AOV-966A	Pressurizer Steam space sample line isolation valve
10.	AOV-966B	Pressurizer Liquid space sample line isolation valve
11.	AOV-966C	"B" loop hot leg sample line isolation valve
12.	AOV-959	RHR loop sample valve
13.	LCV-1003 A&B	1A & 1B Reactor Coolant Drain Tank suction level control
14.	AOV-1600A	Reactor Coolant Drain Tank to Gas Analyzer
15.	AOV-1721	Suction line to Reactor Coolant Drain Tank
16.	AOV-1786	Reactor Coolant Drain Tank to Vent Header
17.	AOV-1787	Reactor Coolant Drain Tank to Vent Header Secondary isolation valve
18.	AOV-1789	Reactor Coolant Drain tank to Gas Analyzer
19.	CV-70	"A" S/G blowdown valves
20.	CV-71	"B" S/G blowdown valves

21.	CV-76	"A" S/G blowdown sample isolation valve
22.	CV-77	"B" S/G blowdown sample isolation valve
23.	AOV-1723	Containment Sump Pump discharge stop valve
24.	AOV-1728	Containment Sump Pump Discharge root valve
25.	AOV-508	Reactor Makeup Water to CV stop valve
26.	CV-74	Instrument air to containment isolation valve
27.	AOV-8418	Demin. water to containment isolation
28.	H_2 Recombiner solenoid valves.	

CONTAINMENT VENTILATION ISOLATION VALVE CLOSURE

29.	AOV-5869	Purge supply outside containment
30.	AOV-5870	Purge supply inside containment
31.	AOV-5878	Purge exhaust inside containment
32.	AOV-5879	Purge exhaust outside containment
33.	AOV-1597	Radiation monitor supply valve
34.	AOV-1598	Radiation monitor exhaust valve
35.	AOV-7970	Containment depressurization valve inside
36.	AOV-7971	Containment depressurization valve outside
37.	MOV-ATV1	Containment air test supply valve
38.	MOV-ATV2 ATV3	Two containment air test vent valve

TABLE 2
PROCEDURE CHANGES

PROCEDURE		CHANGE	PORC APPROVED
PT-2.1	Safety Injection System Pumps	1). Verification by non-test personnel that all safety related valves manipulated during the test have been returned to their required position. 2). Head Control Operator in addition to Shift Foreman is notified before and after testing.	June 4
PT-2.2	Residual Heat Removal System	"	"
PT-2.7	Service Water System	"	"
PT-2.8	Component Cooling Water Pump System	"	"
PT-2.9.1	Check Valve Exercising Quarterly Requirement (RCDT Pump Discharge)	"	"
PT-3	Containment Spray Pumps & NaOH Additive System.	"	"
PT-16	Auxiliary Feedwater System	"	"

TABLE 2
PROCEDURE CHANGES

	PROCEDURE	CHANGE	PORC APPROVAL
M-11.5B	Major Mechanical Inspection of AFWP	Notification of Results & Test Dept. to test redundant pumps before removing a pump from service.	June 11
M-11.5C	Minor Mechanical Inspection of AFWP	"	"
M-11.5E	Pipefitters Inspection of Motor Driven AFWP	"	"
T-41B	Turbine Driven AFWP Removal from Service	"	"
T-41D	Motor Driven AFWP Isolation	"	"

M - Maintenance Procedures

PT- Periodic Test Procedures

T - Turbine Plant Operations Procedures

ATTACHMENT 1

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

LEON D. WHITE, JR.
VICE PRESIDENT

AREA CODE 716 546-2700

January 2, 1979

JAN 3 1979
Copy to JCN, GHP, ELD, GFL, RWM, File

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Containment Purging During Normal Plant Operations
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

This letter is in response to your letter dated November 29, 1978 which was received on December 1, 1978 regarding electrical bypasses and overrides in the containment purge system.

It is our plan to justify limiting purging outlined in option 2 of your letter. Our evaluation for justifying continuation of limited purging during power operation should be completed by July 2, 1979. Pending completion of the NRC staff review of that evaluation the R. E. Ginna Nuclear Power Plant will limit purging to 90 hours per year while the reactor is critical or operating as defined in the R. E. Ginna Technical Specifications. No time restrictions need be placed on purging at shutdown conditions. It should be noted that limiting containment purging to 90 hr/yr result in increased personnel exposure during required Technical Specification surveillances.

A review of the containment ventilation isolation system has been made. The containment ventilation isolation system consists of the four containment purge valves, two containment depressurization valves and two radiation monitor valves. If open, these valves will automatically close on a Safety Injection (SI) signal or on high containment activity. If the containment ventilation isolation system reset is activated while a high containment activity signal or SI signal is present these eight valves could be opened and the automatic closure of these valves is blocked until the reset is deactivated. The reset is deactivated when both the SI signal and the high containment activity signals are cleared.

The purpose of the reset on the containment ventilation isolation system is to allow purging of containment in order to limit

potential hydrogen concentration buildup following a postulated LOCA when high containment activity and SI signals could be present.

Procedures associated with the activation of the containment ventilation isolation system reset have been modified to alert the operator that activating the reset blocks automatic closure of the eight valves on an SI signal. If a high containment activity alarm is present the reset should not be used until the high containment activity alarm has been cleared unless SI has occurred.

A review of all remaining safety actuation signal circuits which incorporate a manual override feature is in progress. This review should be completed by mid February 1979. Until this review is complete the use of bypasses on unreviewed circuitry will be minimized to the maximum extent possible. It may, however, be necessary in certain instances to employ overrides or resets in order to perform certain necessary operations such as instrument tests or equipment maintenance.

Very truly yours,

L D White, Jr.

L. D. White, Jr.

FEB 20 1979

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

TELEPHONE
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February 16, 1979

FEB 21 1979 Copy to JCA (GFL) GHP ECE File

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis L. Ziemann, Chief
Operating Reactor Branch No. 2
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Review of Safety Actuation Circuits with Overrides
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

Your letter dated November 29, 1978 requested restrictions be placed on containment purging during normal operation and that a review of all safety actuation signal circuits which incorporate manual override features be made. The Rochester Gas and Electric Corporation (RG&E) letter dated January 2, 1979 transmitted the RG&E commitments on purging during normal operation and stated that the required review of override features would be completed by mid February 1979. The purpose of this letter is to transmit the results of that review. Details of the review are presented in Appendix A.

The review of all safety actuation signal circuits which incorporate a manual override feature indicates that actuating a particular override (reset) does not cause the bypass of other safety actuation signals. The reset switches described in Appendix A are push button switches located on the control board with no physical restraints.

In all cases where the safety actuation signal is generated automatically and the reset switch is actuated, the safety actuation signal will be inhibited until all logic paths for automatically generating the safety actuation signal have opened. Once all logic paths open, the particular safety actuation signal reset relays de-energize and re-establish the ability to automatically generate the safety actuation signal. Since the reset remains actuated only if the input signals causing the automatic safety actuation signal persist and these input signals are annunciated, no separate annunciation for the reset actuation is necessary. The operator has sufficient information to deduce a certain system is in the reset mode.

In no case does actuation of a particular reset switch prevent the operator from manually operating the equipment from the control board. Therefore, operation of a reset does not prevent equipment from operating which is necessary to mitigate the consequences of a postulated accident.

Very truly yours,

L D White, Jr.

L. D. White, Jr.

LDW:np
Attachment

Appendix A

Review of Safety Actuation Signal Circuits
Incorporating Manual Overrides

The following summarizes the results of a review of safety actuation signal circuits which incorporate a manual override feature. The purpose of the review is to ensure that overriding of one safety actuation signal does not also cause the bypass of any other safety actuation signals:

1. SAFETY INJECTION CIRCUIT:

   This circuit has a reset switch which gives the operator the means of resetting safety injection one minute or longer after initiation. Actuation of the reset switch in itself does not change the state of any equipment, but permits the operator to place the equipment affected by safety injection to the position desired.

   If safety injection is caused by automatic actuation, and the reset switch is actuated, automatic safety injection will be inhibited until all logic paths for automatic safety injection have opened. Once all logic paths open, the safety injection reset relays de-energize and re-establishes automatic safety injection capabilities.

   Manual safety injection initiation is available at all times.

   There is no annunciation of the safety injection circuit being in the reset mode.

   The purpose of the reset switch on the safety injection system is to allow equipment to be realigned for the recirculation phase of a postulated LOCA.

2. CONTAINMENT VENTILATION ISOLATION CIRCUIT:

   This circuit has a reset switch which gives the operator the means of resetting containment ventilation isolation. Once the reset switch has been actuated, most of the equipment will automatically return to the state selected prior to the isolation signal.

   If containment ventilation isolation was caused automatically, either by safety injection or high radiation alarm on containment gas and/or particulate monitors, and this condition continues to exist after the reset switch has been actuated, then containment ventilation isolation cannot be achieved automatically or by the manual isolation switches until this logic clears. Once the automatic logic clears, the containment ventilation isolation reset relays de-energize and re-establishes automatic or manual isolation capabilities.

Manual operation of the valves from the control board is available at all times.

There is no annunciation of the automatic containment ventilation isolation system being in the reset mode.

The purpose of the reset switch on the containment ventilation isolation system is to allow purging of containment in order to limit potential hydrogen concentration buildup following a postulated LOCA when high containment activity and safety injection signals could be present.

3. CONTAINMENT ISOLATION CIRCUIT:

This circuit has a reset switch which gives the operator the means of resetting containment isolation. Once the reset switch has been actuated, some equipment will return automatically to the position selected prior to the isolation signal.

If containment isolation was caused automatically by an automatic safety injection signal, and containment isolation reset switch is actuated without resetting safety injection, containment isolation cannot be obtained by the manual containment isolation switches until safety injection is reset.

Actuation of the reset permits the operator to place the valves affected by the containment isolation signal in the position desired. This capability is necessary so that the operator has flexibility in dealing with post accident conditions within containment.

There is no annunciation of the automatic containment isolation being in the reset mode.

4. CONTAINMENT SPRAY CIRCUIT:

This circuit has a reset switch which gives the operator the means of resetting containment spray. Once the reset switch has been actuated the spray additive tank discharge valves will return automatically to the position called for by the controller prior to the containment spray signal. The containment spray pumps and their discharge valves would require operator action to change state.

If containment spray was caused automatically by the high containment pressure logic, and this logic continues to exist after reset, containment spray cannot be initiated by the manual spray switches. Once the high pressure logic has cleared, the containment spray reset relays de-energize and re-establishes automatic or manual containment spray capabilities.

Actuation of the reset permits the operator to place the valves and pumps affected by the containment spray signal in the state desired. This capability is necessary so that the operator has flexibility in dealing with post accident conditions within containment.

There is no annunciation of the automatic containment spray system being in the reset mode.

5. FEEDWATER ISOLATION RESET:

This circuit has a reset switch which gives the operator the means of resetting the isolation signal to the feed-water bypass valves only. The main feedwater valves will remain closed until the isolation logic clears, and then they automatically assume the position requested by their control circuit.

If feedwater isolation is caused by high steam generator level logic, and this condition still exists after the reset switch is actuated, a safety injection signal would not cause an isolation to that particular feedwater bypass valve. It should be noted that a safety injection signal also causes the main feedwater pumps to be tripped, therefore, closing the feedwater bypass valves on a safety injection signal is redundant.

There is no annunciation of the automatic feedwater isolation system being in the reset mode.

6. NUCLEAR INSTRUMENTATION SYSTEM DEFEAT, BYPASS, AND BLOCK SWITCHES:

This system has several switches which are used for the following purposes:

(a) Defeat Switches - Defeats a permissive which reinstates a trip logic.

(b) Bypass Switches - Bypasses a trip or runback function for calibration or maintenance purposes. Protection is still provided by redundant channel or channels.

(c) Block Switches - Blocks trips generated by source, intermediate, and power range channels. These switches are actuated as permissive setpoints are reached to permit taking reactor critical and up in power. These blocks automatically reset as power is decreased below its particular setpoint.

All the above switches if actuated, are indicated by one or more of the following: status light, alarm on the computer, or actuate an annunciator.

7. INSTRUMENT AND CONTROL DEFEAT SWITCHES:

The following switches and their circuits were reviewed to insure that they are only performing their intended function, and no other safety functions are being bypassed. The purpose for these switches is to be able to switch control from one sensor loop to another for testing, calibration and maintenance purposes. In all cases, reactor trip and safety injection signals are generated prior to defeat switches, and are not affected by switch position.

(a) P/429A Pressurizer Pressure Selector Switch - Used to select two of the four pressurizer pressure channels for controlling pressurizer heaters, sprays, and power relief valve PCV-430.

(b) L/428A Pressurizer Level Selector Switch - Used to select two of the three pressurizer level channels for controlling charging pump speed, letdown isolation, and pressurizer heaters.

(c) T/405E and T/405F Delta T Defeat Switches - Used to defeat a channel from the over temperature and over power turbine runback circuit, and to remove a channel Delta T signal from the input of the summer for generating the average Delta T signal or the Rod Insertion Limit Circuit.

(d) T/401A and T/401B Tavg Defeat Switches - Used to defeat a Tavg channel from the input to the average Tavg summer which is used for full length rod control, condenser steam dump, and pressurizer level setpoint.

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LEON D. WHITE, JR.
VICE PRESIDENT

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March 30, 1979

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Director of Nuclear Reactor Regulation
Attention: Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch No. 2
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Review of Safety Actuation Circuits With Overrides
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Ziemann:

Your letter of November 29, 1978 requested that we perform a review of all safety actuation signal circuits which incorporate manual override features. Our letter of February 16, 1979 provided the results of that review. The purpose of this letter is to supplement and amplify the information on the containment ventilation isolation circuitry which has previously been presented.

The containment ventilation isolation circuitry and the reset function were described in our previous letter. The isolation signal is generated either by safety injection (SI) or high containment radiation alarm on containment radiation monitors. Once a signal is generated, the isolation function is locked in and can only be cleared through use of the reset function, even if the initiating signal has been cleared. Further, once a high radiation signal is generated, the signal itself is locked in and must be cleared. This is not cleared by the containment ventilation isolation reset.

As stated in our previous letter the reset function is not annunciated. However, the signals that generate a containment ventilation isolation signal are annunciated. The position of the purge valves is indicated by lights on the control board. Hence, a combination of an isolation signal (annunciated) and a lack of a close light for the respective valve is positive indication the reset function has been actuated.

The reset switch is a key lock switch and its use is covered by strict administrative controls. Situations in which it might be required are as follows. If a spurious S.I. or high radiation signal were to generate an isolation signal, it would be necessary

to use the switch to clear the isolation signal. Before using the reset, however, the plant operator would clear the spurious signal. Thus, when the reset was employed, it would momentarily block the signals but following release would not block subsequent signals. Procedural precautions alert the operator to the fact that the spurious signal should be cleared prior to using the reset. Strict control of the key for the reset under the Shift Foreman ensures that proper procedures are followed inasmuch as no single operator error can result in improper use of the reset function.

A second situation involving the use of the reset key switch is following the monthly test of the containment ventilation isolation circuitry. In this test, a simulated signal is input into the circuitry. Following completion of the test, the test signal is removed and cleared. Only after this is accomplished is the isolation signal cleared, again under strict administrative controls including decisions by two operators.

A third circumstance which could involve use of the key switch is an actual high radiation signal which isolates containment when purging is desired. Purging could be accomplished by use of the reset function thereby overriding the high radiation signal, however, this is not permitted without a detailed evaluation. In addition, to the best of our knowledge, this has never occurred in nearly ten years of plant operation. The practice, enforced by procedure, in this case is to attempt to clear the high radiation signal in case it is a spurious signal. If it is not a spurious signal, the set point of the monitor would be evaluated and raised, while ensuring that all regulatory requirements for release concentrations (e.g., 10 CFR Part 20 limits) are met. This would permit the high radiation signal to be cleared. Once the high radiation signal were cleared, the ventilation isolation signal could be cleared by momentary use of the reset key switch. Plant procedures for this situation, will explicitly provide information as to the function of the reset function and the need to thoroughly understand and evaluate the situation at hand before using the reset. Again, it has never been necessary, to the best of our knowledge, to use the reset to override an isolation signal in our nearly ten years of plant operation.

Finally, it may be necessary to use the reset function in order to purge containment to limit hydrogen buildup in containment following a design basis loss of coolant accident (LOCA). Following a LOCA, both a high radiation and SI signal will exist. If, based on hydrogen sampling of the containment atmosphere, it is necessary to purge, the plant operator is provided detailed precautions on use of the reset. He is directed to place all valve position controllers in the close position so that no valve will open on initiation of the reset. Then the operator actuates

the reset. Finally, he initiates containment purge. It should be noted that purging to control the post-accident hydrogen concentration is not necessary until at least several days after the event.

In conclusion the Ginna containment ventilation isolation circuitry and procedures regarding its use are adequate. This is based on the detailed procedural controls which have been implemented, the physical control of the reset key switch which involves at least two operators to use, and ten years of successful plant operation.

Very truly yours,

Leon D. White, Jr.

Leon D. White, Jr.

LDW:np

ASSUMPTIONS: Minimum safeguards equipment starts, loss of off site power, 4 loop plant, cold leg breaks.

BREAK SIZE	HEAT SINK	VOIDS	CORE UNCOVERED	PRIMARY PRESSURE
<3/8" (No SI)	S/G through forced or natural recirc.	No	No	Normal
>3/8"<1" (SI-1715)	S/G through natural circulation after 1 day - core boiloff NOTE: System will fill solid	Some in core & T_H	No	Equalizes above S/G relief valve setting
>1"<2" (SI-1715)	S/G through condensation when SI = break flow core boiloff NOTE: S/G can become a heat source	Throughout $T_C > T_{sat}$ no natural circ. due to voids	2" break 1/2 core for ~2 min. PCT-1000°F.	Equalizes below S/G relief setting
>2" (SI-1715)	S/G through condensation initially then core boil-off when SI = break NOTE: S/G becomes a heat source Accumulator dump turns temperature	Throughout	4" break over 1/2 core for ~ 8 min. 6" 80% for 2 min. PCT-1750.	Depends on break size
PORV No heat sink	Core boiloff can't remove the decay heat for the first half hour. NOTE: PZR. LVL. unreliable	Throughout	After 75 min. 3' for 7 min., 1' for 15 min.	Remains at ~1500 for ~1.5 hours
PORV Aux Feed in 30 min.	S/G and Core boiloff NOTE: PZR. LVL. unreliable	Throughout	No	Stabalizes less than secondary relief setting in 50 min.

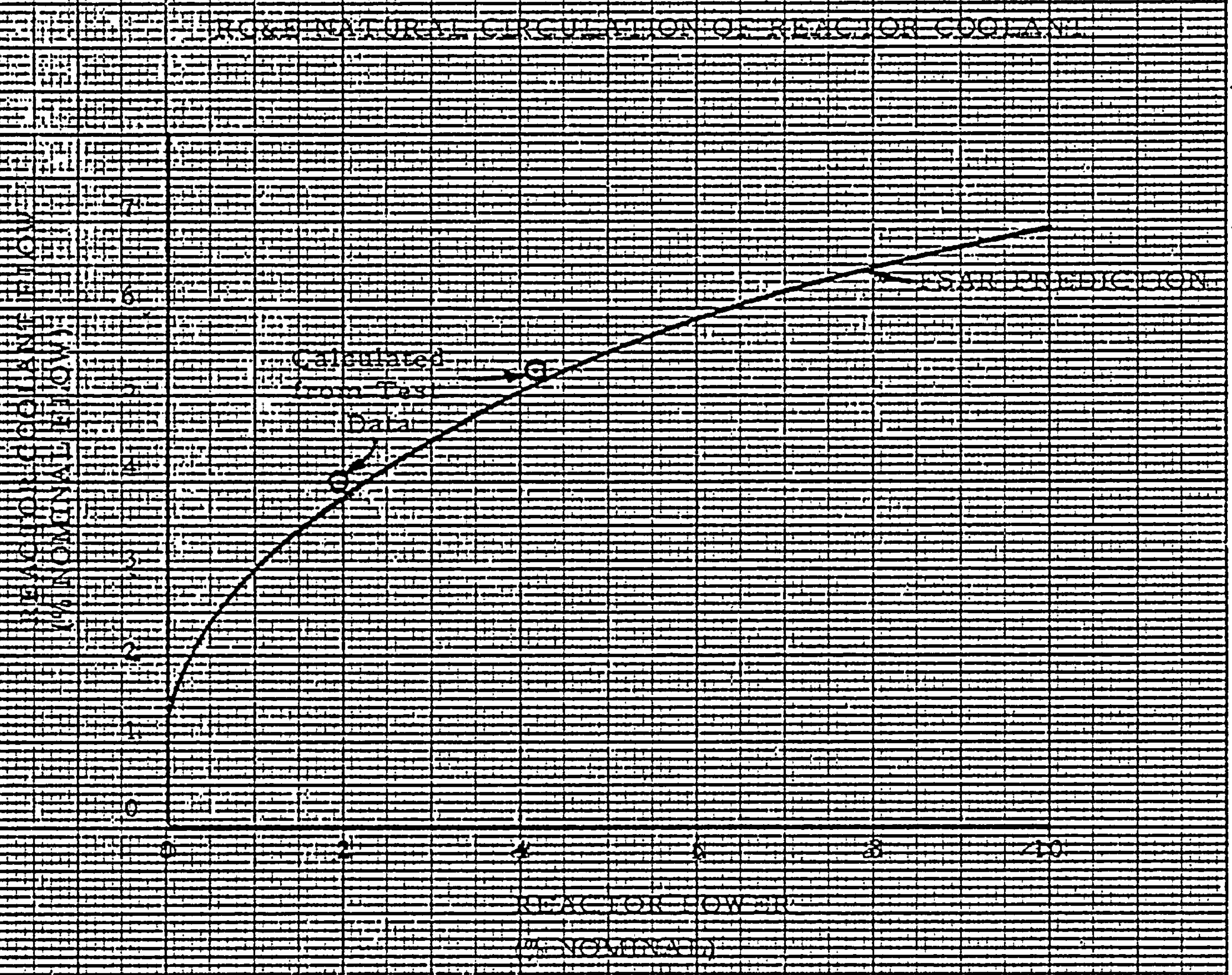
NATURAL CIRCULATION OF REACTOR COOLANT
REACTOR COOLANT FLOW
(% NOMINAL FLOW)
0
1
2
3
4
5
6
7
Calculated
from Test
Data
PSAR PREDICTION
0
2
4
6
8
10
REACTOR POWER
(% NOMINAL)

Figure V-11

ATTACHMENT 4

INITIAL NATURAL CIRCULTION TEST

POWER	TAVG.	ΔT	FLOW
1.7%	550	25.5	4% actual
3.4%	562	40	5.25% actual
5.1%			5.7% projected
6.8%			6.5% projected
8.5%			6.9% projected

[illegible]