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SUBJECT: Forwards responses to SEP Topic Evaluations III-1, "Classification," VII-2, "ESF Sys Control Logic & Design" & VII-3, "Safe Shutdown Sys." Addl comments will be transmitted when available.

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January 23, 1981

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: SEP Topics III-1, VII-2, VII-3
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

Enclosed are the Rochester Gas and Electric responses to SEP Topic Evaluations III-1, "Classification", VII-2, "ESF System Control Logic and Design", and VII-3, "Safe Shutdown Systems", transmitted by NRC letter dated December 12, 1980. Because of the large number (10) of SEP topic evaluations received by RG&E on December 18, 1980, a very detailed review of these topics could not be accomplished. It is expected that additional comments may result from further review of these assessments. These comments will be transmitted as soon as they are available.

Very truly yours,

John E. Maier

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Attachments

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Enclosure 2: RG&E Comments Concerning
SEP Topic VII-2, "Safe Shutdown Systems (Electrical)"

1. In Section 4.1.2, it is stated that one independent auxiliary feedwater "train" consists of the turbine-driven auxiliary feedwater system; the other "train" consists of the two motor driven auxiliary feedwater pumps. This is not correct. The turbine-driven system is a 200% capacity system feeding both steam generators. Each motor-driven system feeds one steam generator. In addition, the standby AFW system consists of two separate trains. Thus, in the Ginna design, there are three 100% capacity auxiliary feedwater trains available to each of the two steam generators.

2. In Section 4.1.2, it is noted that the Standby Auxiliary Feedwater System (SAFS) provides flow in case suction from the CST to the AFW pumps causes AFW pump burnout. The purpose of the SAFS is to provide flow to the steam generators whenever the AFW pumps cannot perform their function, no matter what the cause.

3. In Section 4.1.3, it is stated that the MSIV's fail close upon loss of control air. Although this is true, closure would possibly not occur for several minutes, since the air system would only slowly depressurize to effect closure. However, parallel vent solenoid valves for each MSIV, each powered from a separate battery, would ensure rapid closure in the event of an MSIV closure signal, even in the event of a single failure.

4. Table 4.2 should be revised as noted in the marked-up attachment to more accurately reflect the instrumentation considered by RG&E to be necessary to effect and maintain safe shutdown.

5. As stated in Section 4.1.8, RG&E offsite power system is in compliance with GDC 17, which is considered to take precedence over BTP RSB 5-1 (which has since been superseded by Regulatory Guide 1.139). NRC Safety Evaluations of June 19, 1969, Section 3.7.1 and January 20, 1972, Section 7.1, attested to this compliance Further, a redundant transformer has been purchased and is avilable for use. This transformer can be connected to provide the necessary loads within 7 days.

6. In Section 5.2, it is stated that the Component Cooling Water system does not meet the single failure criterion because of the single discharge line from the CCW pumps through MOV-817. This would result in a loss of cooling water to RCP-1A, RCP-1B, Reactor Support Cooling, and Excess Letdown HX. The same would be true for a failure of check valve 816 to remain open.

   The check valve is normally open at all times. There does not appear to be any credible reason for this check valve to suddenly fail closed. Since this is not a reasonable failure mode postulation, it should not be considered. However, even if the check valve 816, or MOV 817 were to close, it is important to note that no equipment required for safe shutdown would be affected. The only potential problem would be a loss of cooling water to the reactor coolant pump motors.

This issue was addressed in SEP Topic IX-3, "Cooling Water." The information presented during that review will be repeated here.

"NRC Question 5:

How much time is available for operator action between loss of CCW flow to a RCP and pump seizure? What alarms inform the operator of a loss of CCW to a RCP?

"RGE Response 5:

The present RG&E procedure (E-6.1, "Loss of Component Cooling During Power Operation"), specifies that the reactor, and then the reactor coolant pumps, be tripped following a loss of CCW to the reactor coolant pump motors within 2 minutes or before the reactor coolant pump motor bearing temperature reaches 200°F. This is a precaution to prevent any possible pump motor damage due to high temperature. It is not expected that pump motor seizure is of concern until many (greater than 10) minutes following loss of CCW. Even at that time, the RC pump breakers may trip due to high current drain (due to overheating) prior to the pump motor seizure. Westinghouse has performed generic tests to demonstrate that the manufacturer's recommended maximum operating bearing temperature is not exceeded for ten minutes. We have been told that these generic tests do apply to the Ginna reactor coolant pump. Westinghouse also has initiated a generic RCP requalification program, the purpose of which is to demonstrate that an RCP can operate without CCW for 30 minutes without loss of function (i.e. without seizure).

There are a number of alarms and indications which directly measure loss of CCW to the reactor coolant pumps. These include:

" - Alarm A-7, RCP 1A CCW Return High Temperature 125 F or low flow 165 gpm.

- Alarm A-15, RCP 1B CCW Return High Temperature 125 F or low flow 165 gpm.
- High water temperature alarm of 185 F from reactor coolant pump radial bearings.
- Abnormal flow rate indications of F1609 or F1613, in the cooling water return line from either pump.

"Also, additional alarms and indications provide notice of a possible malfunction in the CCW System. These include:

- Indication of pump "off" lights
- Alarm A-13, CCW Surge Tank Low Level 41.2% (LA-618)
- Alarm A-22, CCW Pump Discharge Low Pressure, 20 below normal (PA-617)
- Alarm A-18, volume control tank high temperature 145 F
- Auxiliary A-17, Motor Off, reactor coolant pump or component cooling pump
- Alarm A-23, CCW from reactor support high temperature 150 F

"NRC Question 6:

What are the required operator actions on loss of CCW, SWS, or chilled water system? Are these actions covered in a procedure?

"RGE Response 6:

The most immediate operator actions in response to a loss of CCW (possible alarms and indications given in response 5 above) are to correct the malfunction and restore CCW flow. If this cannot readily be done, the operator is to trip the reactor, and the reactor coolant pumps, as discussed in 5 above. The detailed actions are listed in Emergency Procedures E-6.1 and E-6.2, "Loss of Component Cooling While the Plant is Shut Down." Loss of the Service Water System would also affect CCW to the RCP motors, since the CCW temperature would increase. Emergency Procedures E-38 and E-38.1 prescribe proper operator actions for this condition. Loss of Service Water resulting in heatup of the CCW System would eventually result in some of the above-noted CCW alarms and indications, such as Alarm A-7, RCP 1A CCW Return High Temperature 125 F. Ensuing operator actions will then follow the listing in E-6.1, referenced above."

Based on the available indications and procedures, and the fact that no required safe shutdown equipment is affected by loss of component cooling water, the NRC evaluation should be modified to state that the CCW system meets the intent of all required regulatory criteria, and is therefore acceptable.

7. In Section 5.7, it is noted that MOV-704A provides a path to RHR pump 2. The Ginna arrangement is such that MOV's 704A and B provide parallel suction to RHR pumps A and B, respectively.

8. In Section 5.7, at the top and middle of page 26, it is stated MOV 856 is normally closed, and that MOV's 851A and 851B are normally closed. This is not true. All three of these valves are locked in their safeguards position (open) with power removed.

9. In Section 5.7, page 29, the conclusions are drawn that (1) single failure criterion is not met for MOV's 700 and 701 for RHR inlet and (2) single failure criterion is not met for MOV's 720 and 721 in the RHR discharge. This evaluation is inconsistent with the NRC's safe shutdown evaluation for Ginna, transmitted to RG&E by letter of November 14, 1980. In Section 3.2 of that assessment, the finding is made that, although the RHR system itself is not single failure proof, the alternative methods for attaining cold shutdown are acceptable. The resolution of "RHR System Reliability" in Section 5.1 of that report concluded that the Ginna systems fulfill the requirements for safe shutdown, except for some procedural changes suggested by the NRC. No need for modifications to address the single failure criterion were discussed, or are considered to be necessary.

10. In Section 5.7, conclusion (3) states that the single failure criterion is not met when the RHR system is functioning in the injection phase because of the single inlet line from the RWST to the RHR pumps through MOV 856. This issue has already been addressed and resolved for the Ginna plant. Amendment 7 to Provisional Operating License No. DPR-18 for

Ginna, which includes Change No. 16 to the Technical Specifications, was issued May 14, 1975. Technical Specification 3.3.1.1g states, in part, that A.C. power shall be removed from valve MOV 856 with the valve in the open position. Since this valve is locked in position, no credible single failure will impede the delivery of flow to the RHR pumps from the RWST in the event of a Safety Injection Signal. The assessment should be modified to incorporate this information, noting that the single failure criterion is appropriately met, and that no modifications are required.

11. It is not clear which provision of Regulatory Guide 1.22 is not met by the periodic actuation testing of the Ginna RHR system. In addition to the Safety Injection System test referenced in Section 4.5.1.1.a of the Ginna Technical Specifications, a Diesel Generator Loading Sequence Test is conducted each refueling shutdown. This is described in Section 4.6.1.b of the Ginna Technical Specifications. This test includes actuation of the pumps:

"4.6.1.b Automatic start of each diesel generator and automatic restoration of particular vital equipment, initiated by an actual loss of all normal AC station service power supplies together with a simulated safety injection signal. This test shall be conducted during each refueling shutdown to assure that the diesel-generator will start and following maximum breaker closure times after the initial starting signal for trains A and B will not be exceeded.

	A	B
Diesel plus Safety Injection Pump	20 sec.	22 sec.
plus RHR Pump		
All breakers closed	40 sec.	42 sec.

If additional testing is considered necessary, it should be explicitly detailed, and the basis given. Otherwise, the assessment should be modified to note that the Ginna testing meets the guidelines provided in Regulatory Guide 1.22.

12. In Section 5.7, conclusion (5) appears to be in error. It is stated that the RHR system fails to satisfy BTP RSB 5-1 and Regulatory Guide 1.22 because the RHR isolation valves and their associated interlocks are not tested. Yet this same section of the assessment also quotes the NRC's SEP Safe Shutdown System Review, which concluded that "...this test requirement is not applicable to the Ginna facility, since the interlocks function only when the RHR isolation valves are shut."

    Since this test requirement is not applicable, it is apparent that it need not be met. The assessment should be revised to state that, as noted in the Safe Shutdown Review, the testing requirement for the RHR valves and interlocks is not applicable to the Ginna facility.

13. In the "Summary", it is noted that the "offsite emergency power fails to satisfy the single failure criterion." This is an incorrect summary, in conflict with the conclusions

drawn in Section 5.8. In 5.8, it is stated that "...this design meets the current NRC requirements for offsite power supplied (GDC-17), providing that disconnection of the flexible connections at the main generator terminals can be accomplished within the time constraints imposed by coolant water inventory and battery life, even though this deviates from the guidelines of BTP RSB 5-1." Also, as noted in comment 5 above, a redundant transformer has also been purchased.

The summary section should be modified to note that the Ginna offsite power system meets the current NRC requirements, as stated in GDC-17.

[illegible]

[illegible]

[illegible]

Table 4.2 List of safe shutdown instruments.

Component/ System	Instrument	Instrument Location	Reference
Main Steam	Steam generator level LT & LI 460, 461 and 470, 471	LT Inside Containment LI Control Room*	Dwg. 33013-~~519~~ 544
	Main Steam Pressure PT and PI 468, 469, 478, 479	PT Intermediate Bldg PI Control Room	Dwg 33013-534
Reactor Coolant	Pressurizer level LT & LI, 426, 427, 428, 433	LT Inside Containment Control Room*	Dwg. 33013-424
	Pressurizer pressure PT & PT 449, 429, 430, 431	PT Inside Containment PI Control Room*	Dwg. 33013-424
	RCS temperature TE & TI 409 A & B and 410 A & B	TE Inside Containment TI Control Room	Dwg. 33013-424
Auxiliary Feed	AFWS flow FT 2021, 2022, 2023, 2024 FI 2021, 2022, 2023, 2024	FT Intermed. Build. FI Control Room*	Dwg. 33013-~~519~~ 544
	SAFS flow FT & FI 4084, 4085	FT Aux. Build. Addition FI Control Room*	Dwg. D-302-071-E
Service Water	Pump discharge press. PT 2160 & 2161 PI 2160 & 2161	PT Screen House PI Control Room	Dwg. 33013-529
Chemical and Volume Control	Charging flow FIT 128, FI 128	FIT Auxiliary Build. FI Control Room	Dwg. 33013-433
	RWST level LT 920, LI 920	LT Auxiliary Build. LI Control Room	Dwg. 33013-425

*Indicators are also available at local shutdown panels.

Attachment to Disclosure 1

Table 4.2 List of safe shutdown instruments. (Continued)

Component/ System	Instrument	Instrument Location	Reference
Component Cooling Water	System flow FIT 619	FIT Auxiliary Build. Low flow alarm in control room	Dwg. 33013-436
	Surge tank level LIT 618	LIT Auxiliary Build. LI Control Room	Dwg. 33013-435
Residual Heat Removal	System flow FT 626, FI 626	FT Auxiliary Build. FI Control Room	Dwg. 33013-436
Diesel Generator	Generator output voltage and current	Control Room	
Emergency AC Power	480 V Busses 14, 16, 17, 18, voltage indication	Control Room	
Emergency DC Power	125 VDC Busses 1 and 2 voltage indication	Control Room	

NOT REQUIRED. PROPER OPERATION IS EVIDENT BY PROPER OPERATION OF POWERED EQUIPMENT

Enclosure 2: Comments on SEP Topic VII-2,
"ESF System Control Logic and Design"

Although circuits have been identified which do not contain qualified isolation devices, it has not been determined that "...effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions..." (from GDC-22) will result in loss of the protection function. ESF system design is of sufficient complexity that a detailed design review is required by RG&E in order to properly address the NRC concerns. At this time, we are unable to perform this detailed review. This topic assessment is only one of ten received by RG&E on December 18, all of which required responses on or before January 30, 1981. Although we do intend to respond to the NRC in detail concerning this evaluation, manpower and priority limitations require that this submittal be delayed until after June, 1981.

Type 2 the Executive Vice Pres

Enclosure 3: RG&E Comments on SEP Topic III-1, "Classification"

Comments are provided on the attached marked-up copy of the NRC assessment.

Attachment to Enclosure 3

3. COMPILATION OF IDENTIFIED SYSTEMS

3.1 ENGINEERED SAFETY FEATURE SYSTEMS

The following engineered safety feature systems are required for DBE and safe shutdown:

1. Safety injection system (Emergency Core Cooling System)
   a. High-pressure safety injection pumps
   b. Low-pressure safety injection (RHR) pumps
   c. Passive accumulators
   d. Refueling water storage tank
   e. Boric acid tanks — available but not required
2. Containment air recirculation and filtration system
   a. Fan-cooler units
   b. Charcoal filter units } same thing
   c. Iodine removal units }
   d. Hydrogen recombiner
3. Containment spray system
   a. Containment spray pumps
   b. Refueling water storage tank
   c. Spray additive tank
4. Containment isolation system
5. Containment ventilating system
   a. Recirculation ventilation — same as 2a above
   b. Purge system — alternate to 2d above

Attachment to Enclosure 3

3.2 REACTOR PROTECTION SYSTEMS

The following reactor protection systems (reactor trip channels) are required for DBE and safe shutdown:

1. High nuclear flux (power range)
2. High nuclear flux (intermediate range)
3. High nuclear flux (source range)
4. Overtemperature ΔT
5. Overpower ΔT
6. Low RCS pressure *
7. High pressurizer pressure
8. High pressurizer water level
9. Low reactor coolant flow
10. Turbine trip
11. High containment pressure * (results in SI → Rx trip)
12. Low steam generator pressure (results in SI → Rx trip)
13. ~~Low~~ Low-Low steam generator level *
14. Manual trip
15. Reactor Coolant Pump Breaker Trip

* Only these instruments could be exposed to an adverse environment when required to perform their safety function

3.3 ADDITIONAL SYSTEMS

In addition to the ESF and RPS, the following systems are required for DBE and safe shutdown:

1. Auxiliary feedwater system (Standby)
2. Service water system
3. Component cooling water system
4. Residual heat removal system
5. Chemical and volume control system (only small portion)
6. Offsite power system — NO CREDIT FOR OFFSITE POWER
7. Control room systems — CHARCOAL FILTER CIRCUIT ONLY
8. Emergency power (a-c and d-c) and control power for the above systems and components

— Main steam isolation and relief/safeties

— Pressurizer relief and safeties