ACCELERATED DOCUMENT DISTRIBUTION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9308090126 DOC.DATE: 93/07/27 NOTARIZED: NO DOCKET #
FACIL:50-244 Robert Emmet Ginna Nuclear Plant, Unit 1, Rochester G 05000244

AUTH.NAME	AUTHOR AFFILIATION
MERCREDY,R.C.	Rochester Gas & Electric Corp.
RECIP.NAME	**RECIPIENT AFFILIATION**
JOHNSON,A.R.	Project Directorate I-3

SUBJECT: Provides response to NRC request for addl info re svc water sys.Responses,together w/commitments made & info provided throughout insp period provided sufficient detail to support closure of issues raised during insp.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 40
TITLE: OR Submittal: General Distribution

NOTES:License Exp date in accordance with 10CFR2,2.109(9/19/72). 05000244

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-3 LA	1	1	PD1-3 PD	1	1
	JOHNSON,A	2	2			
INTERNAL:	NRR/DE/EELB	1	1	NRR/DRCH/HICB	1	1
	NRR/DRPW/OTSB	1	1	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	NUDOCS-ABSTRACT	1	1
	OC/LFDCB	1	0	OGC/HDS1	1	0
	REG FILE 01	1	1			
EXTERNAL:	NRC PDR	1	1	NSIC	1	1

R I D S / A D D S

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 504-2065) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 15 ENCL 13

RG&E
ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER N.Y. 14649-0001

ROBERT C. MECREDY
Vice President
Ginna Nuclear Production

TELEPHONE
AREA CODE 716 546-2700

July 27, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-3
Washington, D.C. 20555

Subject: Additional Information-Service Water System
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Johnson:

This letter is in response to the NRC's request for additional information, dated May 24, 1993 regarding the Service Water System. Our responses in Attachment A will 1) indicate how the Service Water System components are modeled in our bounding UFSAR accident analyses to the extent required by previously approved Westinghouse methodologies and our licensing basis, 2) describe steps taken to verify the appropriateness of the SWS hydraulic model, 3) cite closure for previously approved open items from the SWSOPI, and 4) address most other issues as requested in the May 24, 1993 letter. We note, however, that many of the questions are requesting new information, apart from the issues raised during the SWSOPI, and appear to be an extension of the completed inspection involving a reconstitution of the service water design and licensing basis.

In particular, RG&E is concerned about the backfitting implications regarding the staff's questioning of the acceptability of Ginna's licensing basis regarding design for a passive failure. The Ginna Station Service Water System, like that of many other nuclear power plants designed in the 1960's, was designed to perform its safety-related accident mitigation design functions assuming a worst-case single active failure. Failure of a check valve to move to its proper location, when called upon to perform its safety function, was considered a single passive failure, and therefore was not required to be analyzed.

Also, many of the questions appear to be suggesting that additional accident analyses be performed (such as questions 1, 2, 3 and 5) for non-limiting event scenarios. Such new analyses are time consuming, resource intensive and since they are bounded by the limiting analyses, are unnecessary and therefore not consistent with approved Westinghouse methodology and accepted licensing practice. An NRC request for additional analysis or technical justification should be accompanied by valid bases or reasons and should state the purpose of that request. One basis for such a

A001
/11

[illegible]

request could be the results of a PRA, which is not limited to deterministic assumptions regarding single failures. RG&E has not yet completed our PRA, but any sequences resulting in risk levels found unacceptable by RG&E will be addressed. If passive failures (such as referred to question 17) are shown to be a dominant risk outlier, appropriate corrective actions will be warranted.

RG&E maintains that the current two SW pump Technical Specification is correct and safe. This has been demonstrated by analysis (our 9/1/92 submittal). Therefore, we reserve the right, under 10CFR50.59, to remove our 3-pump administrative controls when the circumstances so warrant.

Following our submittals detailing our positions and commitments in relation to GL 89-13 and SWSOPI 91-201, two NRC inspections have been conducted, (Inspection Reports 92-12 dated Oct. 13, 1992 and 93-05 dated March 30, 1993). These inspections reviewed and closed the 3 violations and 3 of the 6 unresolved items. Since these inspections were not referenced in your May 24 letter, it is unclear whether their results were given consideration. The other 3 unresolved items which were not reviewed have been addressed in writing by RG&E, and the commitments made have either been completed and closed by RG&E or were previously acknowledged by the NRC as being acceptably documented and being tracked to closure in the October 13, 1992 inspection report. RG&E has previously supplied information relevant to the specific 91-201 inspection concerns. Those responses, together with the commitments we have made, and the information provided throughout the inspection period have provided sufficient detail to support closure of all issues raised during the inspection.

The closure of inspection items in previous correspondence, the information provided in this submittal, and the commitments we have made to respond to open items provide high confidence in the safety and operability of the Ginna Station Service Water System.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

GJW\291
Attachment

xc: Mr. Allen R. Johnson (Mail Stop 14D1)
Project Directorate I-3
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Ginna Senior Resident Inspector

ATTACHMENT A

Question No. 1

Part 1

Question:
"In order for the staff to judge whether one service water pump can provide adequate flow for the system configuration, the specific SWS flow requirements for each safety-related component must be identified."

Response:
The components that contribute to service water pump demand are not all safety-related; nonetheless, they have been considered relative to operation with one service water pump under various accident conditions. The loads that contribute to service water pump demand are identified on Table 9.2-2 of the Ginna UFSAR. A number of these loads are identified as "critical" to maintain consistency with the term used in the NRC Safety Evaluation Report for SEP Topic IX-3 dated Nov. 3, 1981. As is noted in the UFSAR, these tabulated values were utilized in the design of the plant for sizing of the service water pumps. They do not represent either maximum or minimum flow requirements for those components, that is, they do not represent failure point values. They do represent nominal values that can, with reasonable accuracy, represent expected flow demand on the service water pumps. Since the values were itemized on the original plant component data sheets, they are relied upon as assumptions when evaluating the SW system performance.

Part 2

Question:
"In making this determination, the worst-case conditions must be assumed for each component (not necessarily the design basis accident conditions)."

Response:
Worst case conditions have been assumed relative to accident analyses, and failures assumed which lead to the limiting cases involving parameters such as peak clad temperature, and containment temperature and pressure. Heat removal capability of the containment fan coolers, as discussed in our letters of April 6, 1992 (Action Item 6) and September 1, 1992, determines the adequacy of post-accident containment response, not the number of service water pumps. Service water flowrate contributes to fan cooler capability. Therefore, flowrate was an input in determining the limiting fan cooler capability. Since the fan cooler service water flow and component cooling water heat exchanger flow are significantly higher than the other service water loads during the recirculation phase, applying worst case flow methodology to these

other loads is inconsequential to service water pump capability and the accident analysis.

Emergency operating procedure (ES-1.3) contains the necessary guidance relative to a single service water pump operation post LOCA. Adequate service water flow is established by isolating service water loads not previously isolated automatically, so that a single service water pump would not exceed its runout conditions and so that adequate service water flow is provided to recovery equipment. The procedure provides for isolation of one Auxiliary Building SW loop, one component cooling water heat exchanger, spent fuel pool heat exchangers, and reactor compartment coolers. Containment recirculation fan coolers may be isolated one at a time as is deemed necessary. We note that EOPs cover a realm of events beyond the events that have been considered as part of the plant design basis. Therefore, the EOPs are developed assuming 4 containment recirculation fan coolers are initially available.

RG&E maintains that circumstances which could lead to a single service water pump operating in the recirculation phase post-LOCA (the only event that could lead to recirculation) does not represent the design basis LOCA case in terms of peak clad temperature or the containment integrity. This case has been demonstrated to be bounded by other postulated events, which have previously been analyzed in the UFSAR. The UFSAR was revised (12/92) to include additional information summarizing the various cases considering loss (or no loss) of offsite power and failures which could lead to a single service water pump. UFSAR 9.2.1.4 documents these post accident conditions and analyses have confirmed that the single service water pump case is bounded. This analysis was the basis for our submittal dated September 1, 1992, and supports the existing Technical Specification 3.3.4.

Our analysis assumed 2 (not 4) containment recirculation fan coolers (CRFC) are available, since the worst failure is a diesel generator failure. Therefore, one service water pump has adequate capacity to supply the required recovery load.

<u>Part 3</u>

Question:
"Describe specifically how the flow requirements were determined for each component, including assumptions that were made."

Response:
The service water flows assumed in the analysis were those itemized in UFSAR Table 9.2-2, except for the containment recirculation fan coolers. Since these coolers are important (along with a containment spray pump) in mitigating the consequences of the LOCA with one service water pump, minimum service water flow to these

coolers was utilized considering limiting fouling factors and flow restrictions. This is necessary to ensure that the limiting conditions are analyzed from a containment integrity standpoint. Service water flows to the other loads itemized do not affect the analysis results as explained above. Therefore, the nominal values in the table for each operating component were used. Testing performed on the service water system and calculations performed on the system have confirmed that the flows in this table can be achieved, under either one or two service water pump operation.

Part 4

Question:
"Also indicate to what extent vendor concurrences have been obtained for judgements and evaluations pertaining to equipment performance capabilities."

Response:
The values listed in UFSAR Table 9.2-2 are design values which are based on original plant component data sheets. We utilize these as measures against which testing is judged. The pump manufacturer has supplied information relative to pump runout and NPSH requirements. Limits imposed by RG&E are more conservative than the vendor's recommendations. Service water flowrates for the containment recirculation fan cooling coils and the motor coolers must be revised in the UFSAR as a result of the modification performed to replace the cooling coils during the 1993 refueling outage. This is described in NRC Inspection Report 50-244/93-05 dated March 30, 1993. In summary, since we are utilizing these tabulated values in surveillances or when testing is performed, and they were part of the original design parameters, further vendor supplied concurrences are not obtained. Should values deviate from these, vendor input is part of our overall evaluation process.

Part 5

Question:
"Also, given the flow requirements established for the Component Cooling Water (CCW) heat exchangers, discuss any changes in the ability and times required to shutdown and cooldown the plant."

Response:
The assumed failure modes that would result in only one service water pump operating have significance only during the recirculation phase post-LOCA. During the injection phase, as indicated on UFSAR Table 9.2-2, only one service water pump is assumed to be operating. Given the assumed loss of offsite power component cooling water heat exchangers would be isolated during the injection phase, and are not required for cooling the CCW loads since the source of water for the residual heat removal, safety

injection, and containment spray pumps is the refueling water storage tank (RWST). Following switchover to the recirculation phase post-LOCA, the source of water to the RCS is the containment sump which is at an elevated temperature. Consequently, one component cooling water heat exchanger requires service water, as assumed in the accident analysis, to provide cooling for residual heat removal heat exchangers and safety-related pump cooling. Because of the added service water demand from the component cooling water heat exchanger, the original accident analysis assumed 2 service water pumps operating. However, it was also assumed that all 4 containment recirculation fan coolers would be operating. The worst single failure analyzed in the UFSAR for LOCA is an RHR pump (UFSAR 15.6.4). Therefore, 2 service water pumps can be assumed to be operating in this case. Other failure assumptions, such as those leading to a single service water pump, have been shown to be less limiting and bounded by the existing LOCA analysis (Reference 3, of Question No. 6).

The assumptions that lead to a single service water pump operating during recirculation resulted in 2 containment recirculation fan coolers operating (one 480-V electrical train unavailable from diesel generator failure) although SW flow has been assumed to all four. Under these circumstances a single service water pump has adequate capacity. In addition, the emergency operating procedure ES-1.3 includes guidance on this minimum service water flow case, such that the loads necessary for recovery receive service water flow. Other loads, such as the spent fuel pool heat exchanger, reactor compartment coolers, and non-operating CRFCs and CCW heat exchanger may be isolated if deemed necessary.

Generically, a reduction in service water flow to the component cooling water heat exchanger in the recirculation phase would have the effect of slowing the cooldown during the recovery phase after all limiting conditions have been mitigated. This is inconsequential, since there is no licensing requirement for a specified cooldown rate during recirculation.

For normal operation cooldown (e.g., a refueling outage), use of a single service water pump would be expected to slightly increase the time to achieve cold shutdown. Although not desirable from a commercial standpoint, there are no safety requirements to achieve cold shutdown in any short duration timeframe.

Question No. 2

Question:
"Based on recent test results using the most limiting service water pump (corrected for worse-case lake level, water temperature, instrument inaccuracies, and pump degradation allowed under the inservice testing program), annotate on a simplified diagram of the SWS the pump discharge pressure, flow rates through each safety-related component, flow rates through non-safety-related supply lines (as applicable), and the system configuration/alignment that is necessary to satisfy the flow requirements for safety-related equipment. Also, indicate the electrical division (A, B or non-safety-related) where each component receives power. Describe how this system configuration/alignment will be achieved during an event assuming off-site power is available and also for the condition where off-site is not available, and identify specific time constraints that must be met in order to satisfy all accident analyses assumptions for heat removal. Provide conservative estimates of how long it will take to complete necessary actions for each case (i.e., off-site power available and off-site power not available). The basis for these estimations should be provided, which may include reference to previous plant experience, plant simulations and exercise walk-throughs."

Response:
The attached figures depict on a simplified diagram of the service water supply system the flows to the service water loads for the two cases of interest:

1) Injection phase, loss of offsite power, 1 service water pump
2) Recirculation phase, loss of offsite power, 1 service water pump

Our analysis of various system configurations were conducted using the enhanced service water system hydraulic computer model, Kypipe. The two cases above were examined in order to determine the minimum service water flow to the containment recirculation fan coolers (CRFC) with only one service water pump operating. These values became an input to our accident analysis performed for the one service water pump case reported in our letter dated September 1, 1992.

System runs were made in order to determine sensitivity to the service water header in operation and to produce the least flow through four fan coolers when only one service water pump is operating. For the injection phase the CRFC flow, including the cooling flow to the electric drive motor cooling coils, is approximately 900 gpm. Subtracting the motor cooler flow resulted in a CRFC flow of 840 gpm, the value used in the analysis. (See Question No. 3). Flow values for each of the separate smaller lines, such as safety injection pump bearings, and safety related pump area coolers have not been shown on the diagram individually, since their total does not represent a large load. The flowrates meet or exceed those typical values listed in UFSAR table 9.2-2 for the cases examined.

ATTACHMENT A

The one pump recirculation case included flow to one component cooling water heat exchanger. Consistent with the EOP guidance, flow to the spent fuel pool heat exchanger has been isolated. Although flow to individual CRFCs may be isolated under EOP guidance, all four are depicted in the figure as receiving flow. The average value of 570 gpm per CRFC was used in the accident analysis (See Question No. 3).

Cases have been examined with both loss of and no loss of offsite power and for cases where two CCW heat exchangers were receiving service water flow for the one pump case. In each of these, while service water pump flow did increase to match the system demand, flow was within runout limits and these cases were bounded by the limiting accident analysis cases.

The power sources for the components are listed below.

<u>Motor Operated Valves - Isolate on SI Signal Plus Undervoltage</u>

	MCC	BUS	TRAIN
4663	C	14	A
4773	D	16	B
4609	H	14	A
4780	J	16	B
4616	C	14	A
4735	D	16	B
4615	C	14	A
4734	D	16	B
4614	C	14	A
4664	D	16	B
4670	H	14	A
4613	D	16	B

<u>Motor Operated Valves - Normally Closed When Operating</u>

4027	C	14	A
4028	D	16	B
4013	D	16	B

<u>CRFCs</u>

A	--	14	A
B	--	16	B
C	--	16	B
D	--	14	A

Service Water Pump	Bus	Train
A	18	A
B	17	B
C	18	A
D	17	B

ATTACHMENT A

The primary function of the SW system in accident analysis (equipment cooling excluded) is to provide heat removal by the CRFCs during the injection phase and heat removal from the recirculated sump water during the recirculation phase.

During the injection phase actions are automatic and appropriate time delays are assumed in the accident analysis (See Ref. 1 and Ref. 2). The term "time constraints" implies manual actions which are not required for the injection phase.

The switchover to recirculation is governed by specified RWST level points. The actions required and time available were submitted by Reference 3 and subsequently incorporated into Ginna procedure ES-1.3 "Transfer to Cold Leg Recirculation". The actions are insensitive to availability of offsite power.

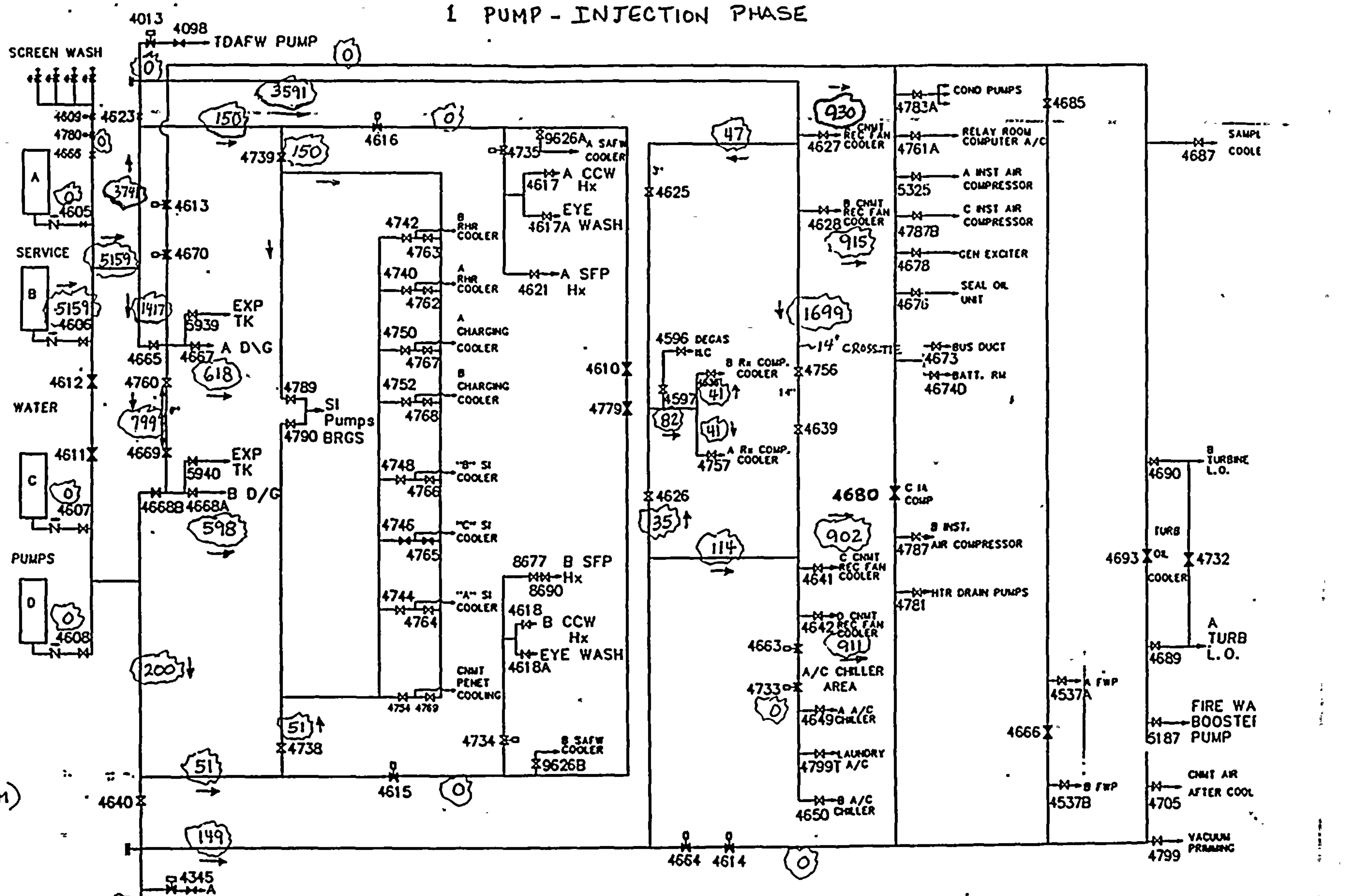
References:

1. RG&E to NRC Request for Amendment to Technical Specifications dated Oct. 16, 1985, Subject - Revise Containment Internal Pressure Limitations, approved under Amendment No. 45 to the Operating License dated August 28, 1991.

2. RG&E to NRC Request for Amendment to Technical Specifications dated Dec. 17, 1992, Subject - Change Required Boron Concentration for Safety Injection and Modeling Requirements for the Boric Acid Storage Tanks.

3. Letter from J. E. Maier (RG&E) to D. M. Crutchfield (NRC), dated June 25, 1982, Subject: SEP Topic VI-7.B, ESF Switchover.

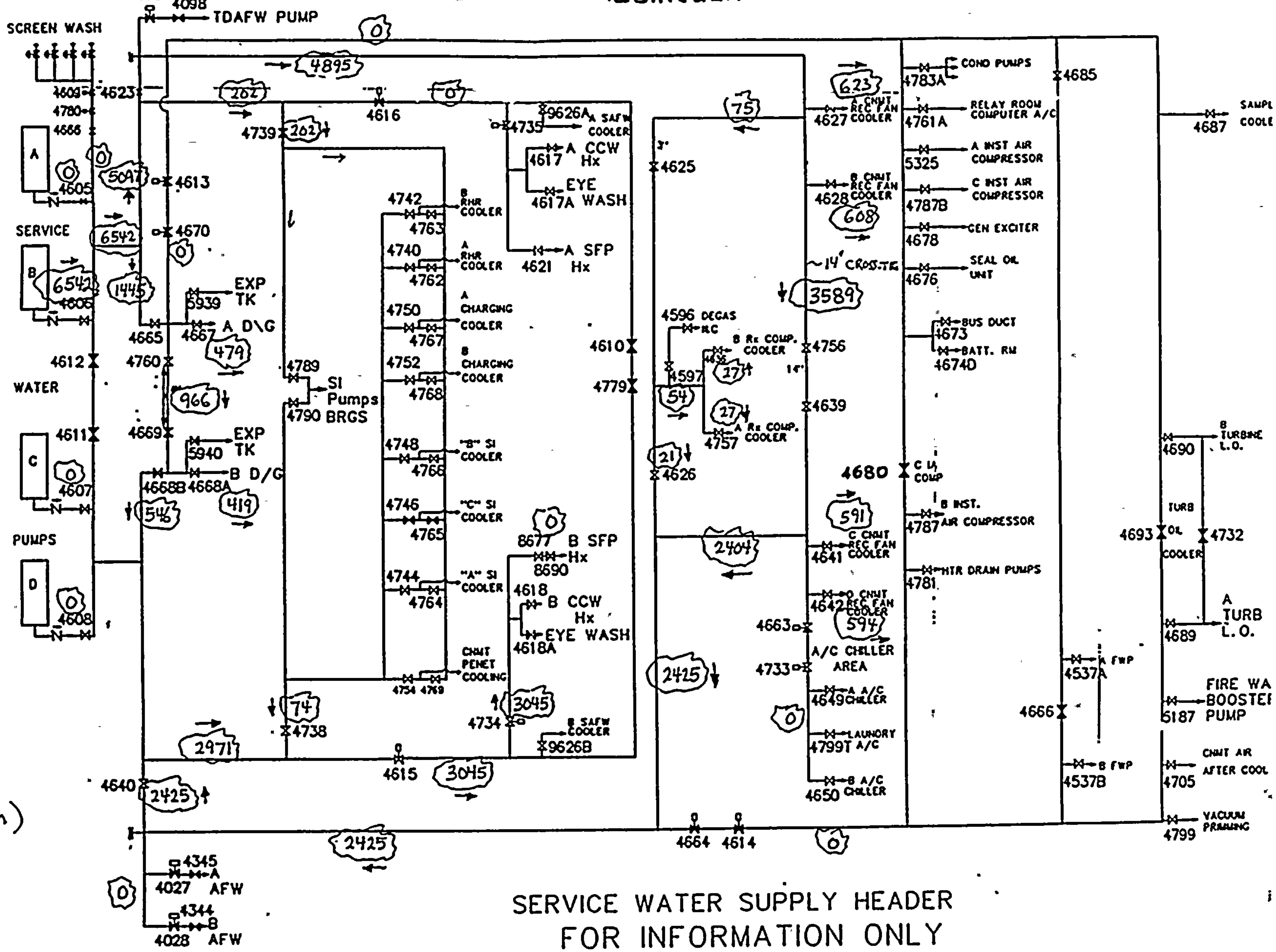
1 PUMP - INJECTION PHASE



SERVICE WATER SUPPLY HEADER
FOR INFORMATION ONLY

KEY
SW FLOW (GPM)

1 PUMP– RECIRCULATION PHASE



SERVICE WATER SUPPLY HEADER
FOR INFORMATION ONLY

KEY

SW FLOW (GPM)

Question No. 3

Question:
"Starting with event initiation, provide bounding assessments (i.e., normal system alignment supplying service water to safety-related components, including the CCW heat exchangers and the spent fuel pool (SFP) heat exchangers, and to non-safety-related components with off-site power available and also for the condition where off-site power is not available) of SWS flow rates over time for single pump operation. The basis for this assessment should be provided, which may include reference to previous plant experience, plant simulations, and exercise walk-throughs. Provide confirmation and vendor concurrence (as appropriate) that SWS pump performance will not be jeopardized by these postulated SWS flow conditions."

Response:
"Event initiation" is assumed to be a UFSAR Chapter 15 events.

The only events in UFSAR Chapter 15 related to the SW system are steam break - containment response and LOCA - recirculation phase. During a steam break SW cools the containment atmosphere through the CRFC(s). During LOCA recirculated sump water is cooled by the RHR heat exchangers via component cooling using SW.

Steam Break - Containment Response

The only parameter in a steam break analysis associated with SW is heat removal via the CRFC(s). A value of 45 MBTU/hr at 286°F was assumed for one SW pump operation. The SW flow required is a function of SW temperature and CRFC air flow. There is no one value of flow which is minimum, since it varies with air flow. A family of curves were developed for this analysis. The value of 840 gpm was utilized in the analysis as discussed in Question 2.

LOCA

The only parameters in the LOCA analysis associated with SW is heat removal via the CRFC(s), and recirculated sump water cooling by the RHR heat exchangers via component cooling using SW. During the injection phase a value of 45 MBTU/hr at 286°F was assumed for one SW pump operation. The SW flow required is a function of SW temperature and CRFC air flow. SW flow of approximately 840 gpm per CRFC was utilized in the analysis.

During the recirculation phase with one SW pump operating, SW flow must be diverted to the CCW heat exchanger. This results in less flow to the CRFC. The average CRFC flow assumed in the analysis was 570 gpm per CRFC. One train of RHR was assumed operating for the one SW pump case. The SW flow to the CCW heat exchanger was assumed to be approximately 2500 gpm. The SW flow values assumed are based on Kypipe simulations of the SW system during the postulated conditions.

ATTACHMENT A

The flow values utilized in the accident analysis are within the capability of one service water pump as established by the hydraulic analyses performed and do not impact the operability of any component.

ATTACHMENT A

Question No. 4

Question:
"Describe periodic surveillances, tests, inspections, and training that will be performed to ensure that the SWS will be capable of satisfying its function during an event and to ensure that all assumptions are valid."

Response:
Periodic inservice testing of the Service Water System includes:

PT-2.7.1: Quarterly service water pump test to verify acceptable flow, differential pressure and vibration.

Quarterly full flow open exercise and prompt closure testing for the service water pump discharge check valves.

Quarterly operability verification of the service water pump selector switches.

Annual service water pump local start verification.

Completion of system/equipment independent verification check.

The PT-2.7.1 acceptance criteria establish limits which assure system performance that meets accident analysis assumptions.

PT-2.3: Quarterly stroke time testing of service water system motor-operated valves.

PT-2.5.4: Quarterly exercise testing of service water system manual valves

PT-2.5.5: Quarterly stroke time testing of service water system air-operated valves.

RSSP-2.1: Safety Injection Functional

RSSP-2.2: Diesel Generator Loading

RSSP-2.4: Annual leak rate testing of the containment recirculation fan coolers service water system valves.

RSSP-2.5: Reverse flow closure test of service water pump discharge check valves.

ATTACHMENT A

RSSP-2.8: Annual leak rate testing of the reactor compartment coolers service water system valves.

Periodic testing of service water system motor-operated valves is controlled by the Ginna Station Motor-Operated Valve Qualification Program (NRC Generic Letter 89-10). This includes refurbishment, maintenance, switch setting and verification and diagnostic testing, both statically and under design-basis differential pressure conditions (baseline).

Inservice inspection of the service water system includes the examination of piping integral attachments per ASME Section XI over the ten year interval and 100% visual examination of safety-related snubbers. Functional testing of safety-related snubbers is also completed for at least 10% of the snubber population.

Relative to training, the plant specific training conducted in the use of the Emergency Procedures and as required by normal operator requalification under the regulations provides the necessary training on all safety systems. RG&E notes that the 91-201 inspection had no unresolved issues relative to operator training. Attention is called to section 3.2.5 of the SWSOPI report noting this area as a strength.

Other training (Maintenance personnel, Health Physics, etc.) is conducted in accordance with our Accredited Training Programs.

Question No. 5

Question:
"Confirm that all existing accident analyses, including safe shutdown/Appendix R analyses, have been updated pursuant to 10 CFR 50.59 requirements to account for single service water pump operation, assuming worst-case conditions of lake level, water temperature, instrument inaccuracies and pump degradation. Similarly, confirm that the current submittal on boron dilution requirements only credits a single service water pump (assuming worst-case conditions) in its supporting accident analyses. Also, discuss specifically the effect of having only one service water pump operable has on the ability to mitigate a steam generator tube rupture event."

Response:
Existing accident analyses consists of a worst case analysis for each accident type. Scenarios that result in single SW pump operation do not represent worst case accidents. Therefore, the existing worst case accidents are not affected by single SW pump operation.

The current submittal deals with boron concentration reduction. The worst case conditions are associated with offsite power available and the worst single failure being a main steam line isolation valve failure. If the single failure was assumed to be one of the two SW pumps vs. a check valve failure the resulting transient would be less severe because containment pressure is more sensitive to the mass/energy released into containment than to a slight reduction in CRFC heat removal due to loss of one of the two SW pumps.

The relationship you have made between one SW pump operation and a SGTR is unclear.

The Steam Generator Tube Rupture (SGTR) event requires operator action for mitigation. The event starts with the tube rupture and requires the operator to identify and isolate the ruptured SG. Next the RCS is cooled down and depressurized. Finally, SI is terminated to stop the primary to secondary leak. No place in this sequence of events is SW required. The SGTR event is insensitive to SW. The only possible relationship is that during RCS depressurization pressurizer steam is released to the PRT. Should the pressure limit of the PRT be exceeded a small amount of steam could be released into containment. Any decrease in CRFC performance due to one SW pump operating would have a negligible effect on containment pressure because of the small amount of steam released and the massive passive heat sinks inside containment.

Single failures are not postulated under Appendix R requirements. The existing Appendix R analysis credits only one service water pump. The analysis also satisfies Appendix R requirements for unmitigated fires in certain plant areas which would render the service water pumps inoperative, such as the screen house.

Question No. 6

Question:
"Provide a detailed description of the assumptions and inputs used in the previous/original limiting containment analyses versus assumptions and inputs used in the current (single service water pump) containment analysis. Provide detailed explanations for any differences of inputs and assumptions between the two analyses. Describe what the service water flow rates are for all components (safety-related and non-safety-related) over time and describe in detail how the containment heat removal rates were determined (including justification for all assumptions) for each of these analyses."

Response:
The original containment analysis is presented in Section 6E of the Ginna FSAR. That section describes the analysis as a design-basis accident assuming 1 spray pump, 2 CRFC(s), 1 RHR pump, and 2 SI pumps. In Reference 1 to this response the staff performed a check calculation using current criteria. The check calculation confirmed the adequacy of the containment analysis with only a slight modification to the temperature profile between 10,000 and 20,000 sec. The profile was increased from 219°F to 250°F. Reference 2 summarizes the current single SW pump containment analysis. As stated in Reference 2 the major assumptions are:

Major Containment Assumptions

Net free volume of containment + sump A + sump B =	1.0×10^6 ft^3
Initial pressure	15.7 psia
Initial temperature	120°F
Initial humidity	100%
Initial outside temperature	100°F
Containment Recirculation Fan Cooler	
number of CRFC operating	2
heat removal during injection phase	Table 1A
heat removal during recirculation phase	Table 1B
start 2 CRFC	42.5 sec.
Containment Spray	
number of spray pumps operating	1
flow rate per spray pump	1300 gpm
RWST water temperature	80°F
spray setpoint with uncertainty	32.3 psig
containment pressure to stop spray	≤4 psig
Recirculation Phase	
start recirculation, RWST level	≤28%
RHR heat exchanger performance (calculated for CCW flow = 1500 gpm @ 115° sump flow = 1550 gpm nominal)	Table 3
Passive heat sinks	Table 2

ATTACHMENT A

Major Mass/Energy Input Assumptions

Design - Basis LOCA mass/energy input	Ref. 2
Reactor vessel volume	2525 ft^3
Decay Heat + 20% until 1000 sec. + 10% after 1000 sec.	Ref. 3
Stored energy in fuel and RV simulated by additional	10% decay heat

The single SW pump analysis assumptions are consistent with the original analysis as to the number of spray pumps, number of CRFC, and number of RHR pumps operating. The other assumptions are consistent with assumptions used in analysis submitted to the staff by Reference 5 and 6 and in some cases represent more conservative assumptions.

The SW flow rates assumed in the analysis are consistent with the discussion presented in response to Question #3.

Heat is removed from containment by the CRFC(s), containment spray, passive heat sinks, and sump water during recirculation via the RHR heat exchanger.

CRFC(s)

Heat is removed from the containment atmosphere based on a table of heat removal vs. temperature. The table is generated for minimum heat removal capability of the CRFC.

Containment Spray

The computer code ("Gothic" developed by EPRI) calculates heat removal by the spray based upon the interaction of the drops and containment atmosphere. The heat removal rate is dependent upon drop diameter, drop temperature spray flow rate, and atmospheric conditions.

Passive Heat Sinks

Heat transferred to the passive heat sinks is calculated by the computer code. The heat transfer coefficient specified is one times Tagami until the end of blowdown followed by an exponential decrease to Uchida. This is standard methodology used in NRC accepted containment integrity analysis.

RHR Heat Exchanger

During recirculation heat is removed from the recirculated water by the RHR heat exchanger. The heat exchanger model in the computer code is tuned to remove less heat than was calculated for the actual heat exchanger with CCW flow = 1500 gpm @ 115°F and RHR flow = 1550 gpm @ sump temperature.

References:

1. NRC letter from D. M. Crutchfield to J. E. Maier, dated Nov. 3, 1981, Subject: SEP for Ginna, Evaluation Report on Topics VI-2.D and VI-3.

2. RG&E to NRC letter from R. C. Mecredy to A. R. Johnson, dated Sept. 1, 1992, Subject: SWSOPI-Response to Deficiency 91-201-08.

3. RG&E to NRC Request for Amendment to Technical Specifications dated Oct. 27, 1987, Subject - Revise Core Safety Limit Curves for Increase in SG Tube Plugging, approved under Amendment No. 25 to the Operating License, dated February 23, 1988.

4. Standard Review Plan, NUREG 75/087, Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling".

5. RG&E to NRC Request for Amendment to Technical Specifications dated Oct. 16, 1985, Subject - Revise Containment Internal Pressure Limitations, approved under Amendment No. 45 to the Operating License, dated August 28, 1991.

6. RG&E to NRC Request for Amendment to Technical Specifications dated Dec. 17, 1992, Subject - Reduce Boron Concentration for Safety Injection.

ATTACHMENT A

TABLE 1A: CRFC Heat Removal - Injection Phase

Temperature °F	Heat Removal BTU/sec	Temperature °F	Heat Removal BTU/sec
80	0	200	4229
210	4628	220	5314
230	6862	240	8139
250	9176	260	10133
270	11091	280	11969
290	12847	300	13484

TABLE 1B: CRFC Heat Removal Recirculation Phase

Temperature °F	Heat Removal BTU/sec	Temperature °F	Heat Removal BTU/sec
80	0	200	3666
210	4011	220	4607
230	5947	240	7055
250	7954	260	8784
270	9614	280	10375
290	11136	300	11689

ATTACHMENT A

TABLE 2: Passive Heat Sinks

	Description	Heat Transfer Area ft^2	Material	Thickness in.
1.	Insulated Containment Wall and Dome	36,285	stainless steel insulation steel concrete	0.019 1.25 0.375 42
2.	Uninsulated Containment Dome	12,370	steel concrete	0.375 30
3.	Sump A Walls and Floor	2,317	steel concrete	0.25 36
4.	Sump B Walls and Floor	299	concrete steel concrete	24 0.25 19.56
5.	Compartment Walls	18,846	concrete	17
6.	Inside Refueling Cavity	6,752	stainless steel concrete	0.25 24
7.	Structure on Operating Floor	3,752	concrete	1.2
8.	1.5 inch steel	7,120	steel	0.75
9.	1.0 inch steel	1,150	steel	0.47
10.	0.5 inch steel	9,902	steel	0.29
11.	Grating	14,000	steel	0.063
12.	Basement Floor	6,576	concrete steel concrete	24 0.25 19.56
13.	Intermediate Floor	9,672	concrete	3
14.	Operating Floor	11,818	concrete	12
15.	Outside Refueling Cavity	6,132	concrete	29.5
16.	Steel	6,219	steel	0.816

ATTACHMENT A

Thermophysical Properties of Passive Heat Sinks

	Density lb/ft^3	Conductivity BTU/hr ft°F	Specific Heat BTU/lb °F
Insulation	6.67	0.0208	0.3
Concrete	141	0.73	0.21
Steel	489	30	0.111
Stainless Steel	496	15	0.11

ATTACHMENT A

TABLE 3

RHR Heat Exchanger Performance

Sump Temperature °F	Heat Removal BTU/hr
160	4,296
180	6,232
200	8,184
220	10,154
240	12,122
260	14,092

Question No. 7

Question:
"Confirm that for a steam line break event at hot zero power with off-site power available, that one service water pump is sufficient to mitigate the event. Similar to item 6 (above), provide a detailed explanation (justification) of any differences in assumptions and inputs used in the previous/original limiting analysis versus assumptions and inputs used in the current (single service water pump) analysis. Also, describe what the service water flow rates are for all components (safety-related and non-safety-related) over time and describe in detail how the containment heat removal rates were determined (including justification for all assumptions) for each of these analyses."

Response:
From Reference 1 & 2 the most limiting single failure associated with a steam line break at HZP, offsite power available is a failure of one spray pump. If the single failure is changed from a spray pump to a SW pump, SW flow to the CRFC(s) would be reduced. The reduction in SW flow would result in a reduction in heat removal by the CRFC(s). However, the reduction in heat removal would be far outweighed by the additional heat removal of the second spray pump. Therefore, one SW pump is sufficient to mitigate the event and is bounded by the limiting case analyzed.

The only assumptions that would change between the two cases is the input for spray flow and the input for CRFC heat removal vs. temperature. All other inputs are identical.

References:

1. RG&E to NRC Request for Amendment to T.S. dated Oct. 16, 1985, Subject - Revise Containment Internal Pressure Limitations, approved under Amendment No. 45 to the Operating License, dated August 28, 1991.

2. RG&E to NRC Request for Amendment to T.S. dated Dec. 17, 1992, Purpose - Reduce Boron Concentration for Safety Injection

ATTACHMENT A

Question No. 8

Question:
"Describe actions that have been taken to validate the SWS flow model for various configurations of split and cross-connected system operation that may be encountered and, in particular, discuss validation of the flow model with respect to single pump operation for various system operation configurations."

Response:
NUS Corporation prepared a hydraulic model of the Ginna Station Service Water System. The NUS model was provided as calculation 9608-M-01 dated 2/88 which was used in conjunction with the University of Kentucky Computer Code, KYPIPE, Fortran Version 3.2, 4-5-88 to provide preliminary engineering analysis for EWR 1594. This EWR included the installation of a larger capacity Spent Fuel Pool Cooler as part of this Service Water System. At completion of the cooler installation in 1989, the model was not validated and As-Built.

As part of the NRC Service Water System Operability Inspection during 12/91, RG&E committed to "review and enhance the analytical modeling and capability of its existing Service Water hydraulic model so that an analytical basis and test specification for the rebalance of the entire Service Water System is available prior to the 1993 Refueling Outage". RG&E Design Analysis, DA-ME-93-0044 dated 3/93 documented this comprehensive review, update and validation of the Service Water System Hydraulic Model. This analysis included the following:

1. System P&ID markups were produced to clearly define the model boundaries relative to current "As-Built" condition.

2. The NUS model input sheets were reviewed against the Seismic Upgrade Isometrics. All discrepancies such as line lengths, pipe diameter, number of fittings, etc. were incorporated in updated model data sheets. This review constituted approximately 100% of the "Safety Related" portion of the system.

3. 10% of the non-safety related portion of the system was reviewed relative to the NUS data input references, other available references and actual walkdown information and appropriate changes into the model input sheets were made.

4. The model was updated to show separate lines for the Containment Air Recirculation Coolers and the Containment Air Recirculation Fan Motor Coolers.

5. The model was also updated to show as separate lines the Diesel Generator Jacket Water Coolers and the Lube Oil Coolers.

6. The model's Service Water Pump performance output was reviewed based on current pump test data and was verified acceptable.

7. The model's pipe line roughness parameters were changed to be consistent with current reference (Cameron Hydraulic Data) for "Old" pipe as it was felt that the 2.67 multiplier used by NUS was ultra-conservative based on current test and operating data.

8. The minor head loss coefficients included for the model line updated input data sheets were evaluated against recent test data. Changes were made to the model input data as appropriate.

9. A final overall operational evaluation was performed relative to specific model outputs during two (2) pump normal operations. Additional adjustments were made to the model inputs to further simulate actual plant parameters to the extent practical.

This upgrade of the model provides a model which can be used to evaluate system flow and pressure for nearly all system configurations and pump combinations (including single pump operation). The use of Service Water to supply the suction source of water for the Auxiliary Feedwater Pumps and the Standby Auxiliary Feedwater Pumps has not been modeled.

ATTACHMENT A

Question No. 9

Question:
"Provide the results of the check valve single failure analysis using the validated flow model, including SWS header pressure and SWS flow rates through the failed check valve and all other components assuming that the two most limiting service water pumps are operating. Justify all assumptions. Also, describe the results of the Ginna probabilistic risk assessment (PRA) with regard to failure of a SWS pump discharge check valve, including justification for check valve failure frequencies that were assumed."

Response:
The analysis of the check valve failure was documented in NSL-0000-DA043 approved 9/14/92. The results were reported in our response dated 4/6/92 in Attachment 2, Action Item 2. The analysis was not repeated using the current revision of the hydraulic service water model. Based on studies performed using the version of the model on which the analysis was based and the current version of the model, the flow values would not be expected to be significantly different. The conclusion would remain the same, that is, the flow directed to service water loads with a stuck open check valve in a 2 pump configuration exceeds the flow resulting from a single service water pump. Since analysis has shown the latter to be acceptable and bounded by the design basis event, the check valve failure postulated is also bounded. As discussed in our 4/6/92 response, the postulated check valve failure is a passive failure and beyond our current licensing basis. The PRA discussion is included in the cover letter.

ATTACHMENT A

Question 10

Part 1

Question:
"Describe the periodic maintenance, surveillance, test and inspection activities that are performed that provide assurance that a SWS pump discharge check valve will function properly."

Response:
This requested information was supplied in Question No. 4. The maintenance program was reviewed during the SWSOPI and covered in Section 3.3 of the 1/30/92 NRC report.

Part 2

Question:
"Also, describe how the failure of a pump discharge check valve to function will be identified and addressed by the operators, including a description of training that has been provided and periodic training that will be provided in the future."

Response:
Following a concern being identified by the NRC in the SWSOPI, RG&E instituted interim measures to alert operators of this postulated failure and identified mitigating actions. A heightened awareness was provided to operations staff initially through the addition of this information in the operators plan-of-the-day. Technical analyses were initiated and subsequently completed and formalized demonstrating that the passive failure of a discharge check valve is bounded by conditions leading to single service water pump operation.

Changes have been made to procedure AP-SW.1, Service Water Leak, to specifically address this postulated failure. This procedure is entered when responding to a service water leak, service water header pressure low alarms, and other specified entry conditions. Operators verify at least one service water pump is running in each SW loop and are cautioned that abnormally low SW loop pressure may indicate an idle pump check valve is open. Instructions are provided for indications that SW pressure is 1) approximately equal in both loops, 2) SW pressure less than 40 psig in either loop, 3) and pressure in both loops depressurized. Operator training on this failure is covered under the training specified for Abnormal Procedures. (See also Question No. 16)

Part 3

Question:
"Provide confirmation that operator action is sufficient to ensure that a SW pump discharge check valve failure will remain bounded by the accident analyses that have been performed."

Response:
As discussed in our responses to the SWSOPI unresolved item 91-201-07 provided as Action Item 2 of our letter dated April 6, 1992 and with Attachment A of our letter dated September 30, 1992, a discharge check valve failure to close is bounded by the single active failure of a diesel generator. In the latter situation, the flow from only one service water pump has been assumed to occur. It was demonstrated and reported in our letter dated September 1, 1992 that the single service water pump operating post LOCA does not produce the limiting conditions which challenge containment integrity.

In the analysis of the discharge check valve failure, it was determined that, because of the system alignment which would occur in this case, an effective service water flow greater than the design flow from one pump would occur; hence the one service water pump case bounds that analysis. RG&E plans to include the results of the discharge check valve failure in the next annual (December 1993) update of the UFSAR. Ensuring that bounding analyses remain bounded is an examination that occurs during the 10CFR50.59 process as plant changes are proposed. A proposed plant modification which would alter system flow rates would require validation that the probability or consequences of existing accidents and malfunctions described in the UFSAR are not increased.

ATTACHMENT A

Question No. 11

Question:
"Provide the conclusions and recommendations of the revised single failure analysis. Discuss corrective actions that are being taken to address vulnerabilities. Also, confirm that the revised single failure analysis encompasses all operating configurations of the SWS that are allowed and include a description of the specific configurations that were considered to be applicable in this regard."

Response:
The results of our revised single failure analysis (still in the acceptance process) revealed that the Ginna Station SWS, as designed and installed, has adequate and sufficient redundancy of components and administrative controls to ensure the system can perform its intended safety-related functions with a single active component failure during both accident conditions and normal power operation. Consequently, no corrective actions were initiated as a result of the single failure analysis as there were no vulnerabilities identified.

The specific operating configurations addressed in the analysis were:

1. LOCA without Loss of Offsite Power
2. LOCA with Loss of Offsite Power
3. Post-LOCA Recirculation Phase operation
4. Auxiliary or Standby Auxiliary Feedwater Operation taking suction from the service water system.

Question No. 12

Question:
"Identify which safety-related components do not need service water cooling during the injection phase of an accident and provide justification for this position, including vendor concurrence. The justification should include consideration for small break loss of coolant accident (SBLOCA) conditions, where pumps may be required to operate at or near shut-off head conditions. Provide a conservative estimate for when service water cooling must be restored to this equipment and describe how this will be accomplished. The basis for these estimations and actions should be provided, which may include reference to test data, previous plant experience, plant simulations and exercise walk-throughs."

Response:
Components that require SW during the injection phase of an accident are supplied with SW as the valves in these lines are open. Components that are non-essential are isolated if offsite power is lost. Components supplied with service water during the injection phase are shown in UFSAR Table 9.2-2 and the critical loads are currently marked with a footnote "a". However, analysis has shown that service water is not essential to be supplied to the pump area coolers as acceptable environmental conditions will still be maintained post accident (UFSAR 9.4.2.4.1, 9.4.9.4, 3.11.3.2.1, 9.4.2.2.3). In addition, we also believe that, although SW is supplied to the safety injection pump bearing water jackets and the valves in these lines are locked open, the pumps could be expected to operate without immediate danger without the external source of service water. Service water is expected to be necessary during the sump recirculation phase. We note that for small break LOCA events where RCS pressure remained near the discharge pressure of a safety injection pump, a minimum recirculation flow of 100 gpm would continue through the pump, representing 25% of the flow at the best efficiency point, negating any increased axial thrust load effects which could be expected at very low pump flow. For the SBLOCA event (4" break) analyzed in the UFSAR, where RCS pressure remains at 1200 psia for the first 200 seconds before dropping (UFSAR Figure 15.6-21), safety injection pump flow would increase to over 200 gpm. The pumped fluid would be cool (60-80°F) throughout most of the injection phase. Therefore, an external source of service water cooling is not essential for the accident cases analyzed in the UFSAR, although does provide additional safety margin for the case considered below.

For a postulated SBLOCA of sizes less than 4" (not currently limiting and not analyzed), RCS pressure may remain closer to the discharge pressure of the pump. In that case, a 100 gpm minimum flow would exist, however, the pumped fluid (145°F - 175°F water) from the boric acid storage tanks would be pumped for a longer period of time than the 4" or larger breaks (perhaps 10-20 minutes). RG&E has assessed this condition and believes that there is adequate assurance that the pump could operate continuously without excessive thrust bearing heatup because:

a) the duration while operating on higher temperature boric acid is relatively short (10-20 minutes)
b) the temperature rise across the pump is low (10°F rise)
c) oil viscosity and bearing temperatures of up to 190° can be tolerated.
d) the bearing housing design provides an adequate heat sink
e) unbalanced axial thrust load is low even at 100 gpm recirc flow
f) heat transfer across the shaft to the bearing area is small
g) similar pumps used in more severe applications without external cooling exist in commercial and Navy applications.

Nonetheless, RG&E is not proposing to remove service water cooling for the bearings on the safety injection pumps during the injection phase.

The only equipment that require SW to be restored is the CCW heat exchanger prior to initiation of recirculation. The basis for recirculation is discussed in the response to Question No. 2.

ATTACHMENT A

Question No. 13

Question:
"Similar to item 12 (above), provide justification and vendor concurrence for the position that component cooling is not required during the injection phase of an accident for the RHR pump mechanical seal coolers and bearing water jackets, the safety injection pump mechanical seal coolers, and the containment spray pump mechanical seal coolers."

Response:
The current configuration of cooling water lines supplied to the components listed above is as originally designed, and its design basis has not changed. Cooling water for the pumps listed above is supplied by the component cooling water (CCW) system (UFSAR Figure 9.2-4) not the service water system. Water from the CCW system is supplied to the individual mechanical seal coolers for safety injection (SI) pumps, containment spray (CS) pumps and residual heat removal (RHR) pumps, and directly to the bearing water jackets for RHR pumps when CCW pumps are operating. Pumped fluid is recirculated through the mechanical seal coolers and returns to the pump, directed at the mechanical seal rotating element. Valves on the branch lines are locked open.

The component cooling water system serves as an intermediate system between the radioactive fluid system and the service water system, reducing the probability of radioactive fluid leakage to the environment via the service water system. Component cooling water circulates into various components where it picks up heat from other systems and transfers the heat to the service water system via the component cooling water heat exchanger. During the injection phase post-LOCA service water is isolated from the tube side of the CCW heat exchangers. (Component cooling water flow to the RHR heat exchangers is unnecessary during injection, since the source of water for the RHR loop is RWST water). This supports the design basis for the service water system that only one service water pump is assumed during the injection phase, since the CCW heat exchanger is a significant service water flow load (5070 gpm in UFSAR Table 9.2-2). Component cooling water pumps are shed on a safety injection plus undervoltage signal, thus CCW is not passed through the individual seal coolers for the pumps listed above during the injection phase.

Since the pumped fluid is cool (60° - 80°F) from the RWST during the injection phase and circulates from the pump and back to the seal faces, there is no need for the CCW. For SBLOCA (less than 4") safety injection pump fluid may be as high as 175°F from the boric acid storage tank (BAST). For similar reasons (as Question No. 12) the mechanical seals are not greatly stressed, since the duration is short and flashing across the seal faces will not occur. Following pumpdown of the BAST, the pumped fluid transfers to much cooler (60° - 80°F) RWST water. Hence, the CCW fluid is not essential in providing cooling for the mechanical seals during the injection phase.

ATTACHMENT A

The CCW flow to the RHR bearing water jackets is necessary only during the recirculation phase. Providing external cooling for an oil lubricated bearing design for these relatively slow (1770 RPM) speed pumps when the pumped fluid is cool (60-80°F range from the RWST) is not essential. Since the RHR pumps take suction from the containment sump during recirculation, however, and the pumps are upstream of the RHR heat exchangers, cooling for the bearings is warranted. Note that the design of containment spray pumps which operate at 3550 RPM but are downstream of the RHR heat exchangers, do not require bearing cooling for the pumps oil lubricated bearings at any time. The pumped fluid would be either RWST water (injection phase) or water recirculated from the containment sump (recirculation phase) after having been cooled by passing through the RHR heat exchanger. The RHR pump manufacturer has confirmed that operation without external cooling for the bearings is not essential when the pumped fluid is 120°F or less. Therefore, bearing oil cooling for RHR is warranted only during sump recirculation when pumped fluid temperature is elevated.

Question No. 14

Question:
"Describe how long the Emergency Diesel Generator (EDG) coolers have been in service and what periodic inspections and preventative maintenance activities are typically performed relative to these coolers, including frequencies."

Response:
The Ginna EDG jacket water heat exchangers and lube oil heat exchangers are part of the original plant equipment. Therefore, they have been in service approximately 24 years.

Currently, the subject coolers are inspected annually as required by Ginna maintenance procedures M-15.1, "A or B Diesel Generator Inspection and Maintenance" and M-37.164, "Clean and Inspect Diesel Generator Lube Oil and Jacket Coolers." These procedures require a specific inspection for bivalves (clams and zebra mussels). Eddy Current inspections are also performed on the heat exchanger tubes and tubes are plugged as required. Adequate flow through the coolers is verified by observing acceptable jacket water and lube oil temperatures during the monthly/annual Technical Specification operability tests. Procedures PT-12.1, PT-12.2, RSSP-2.2, RSSP-2.3 "A" and RSSP-2.3 "B" are the Ginna procedures associated with the flow testing. In addition, pressure drops across the coolers, jacket water temperatures and lube oil temperatures are monitored and recorded every four (4) hours as part of the turbine building logs. The continuous monitoring of these parameters would provide an indication of debris buildup, fouling or loss of cooler performance. Unscheduled maintenance is performed based upon any unacceptable results obtained from this monitoring.

Comprehensive baseline pilot testing of the coolers has been completed during the summer of 1992. This testing determined that the Jacket Water and Lube Oil coolers have sufficient design margin for accident requirements. New instrumentation was installed during the 1993 Refueling Outage in preparation for the first annual test scheduled for the summer of 1993. Long term thermal performance test frequencies for the coolers will be established based on the results of the first three years of annual testing. This testing program is consistent with the requirements of Generic Letter 89-13 and will be formally performed and documented in Ginna PT-60 series procedures.

ATTACHMENT A

Question No. 15

Question:
"Assuming the maximum decay heat load in the spent fuel pool, describe how long service water to the SFP heat exchanger can be isolated, including supporting justification. Also, describe alternative measures that can be taken that are included in emergency procedures for cooling the spent fuel pool during an event."

Response:
The maximum spent fuel pool (SFP) heat load occurs during a full core discharge. Since there is no fuel in the reactor vessel an event in the RCS is meaningless during this time. Description of cooling to the SFP during maximum heat load has been discussed in Reference 1 and approved by the NRC in Reference 2.

The maximum SFP heat load during normal operation when only stored fuel is in the SFP (not during full core discharge) is approximately 4.17 MBTU/hr (Ref. 1). From Reference 3 the time to heatup from 150°F to 180°F is approximately 14 hours. This has been considered to be sufficient time to provide alternative cooling should some event require isolation of SW to SFP.

Reference:

1. RG&E to NRC letter from J. E. Maier to D. M. Crutchfield dated June 9, 1981, Subject: Spent Fuel Pool Cooling System; SEP Topic IX-1

2. NRC to RG&E letter from D. M. Crutchfield to J. E. Maier, dated Nov. 3, 1981, Subject: SER for Proposed Spent Fuel Pool Cooling System

3. RG&E Design Analysis, NSL-0000-DA-030, Rev. 0, approved 3/12/91

ATTACHMENT A

Question No. 16

Question:
"Define the SWS header low pressure setpoints for all system configurations and discuss the basis for the setpoints that have been established."

Response:
Our response to this question, 91-201-12, was updated in our 9/30/92 submittal. We stated that the 40 psig value in AP-SW.1, "Service Water Leak", is sufficient under accident or transient conditions to ensure that adequate service water flow would be provided to the required service water loads. An analysis was formalized (DA-ME-92-0116 approved 12/21/92) which documented recommended alarm limits and operator action limits for two and three pump operation and provides the basis for the settings. Plant data on the service water pressure has been trended to validate the calculated values. Interim instructions were provided to the plant operations staff by the engineering department prior to the implementation of procedure changes. Plant procedure changes are in process to reflect the following settings:

2 pump operation

55 psig	Administrative Limit
50 psig	Computer Alarm Setting: verify one pump in each loop operating; continue leak inspection; verify flow to CCW heat exchangers; check operation of temperature control valves; initiate SW loop separation if necessary.
45 psig	start 3rd SW pump; initiate controlled shutdown if pressure not stabilized
40 psig	trip the reactor; proceed to E-0

3 pump operation

60 psig	Administrative limit
55 psig	Computer alarm setting; verify 3 pumps operating; continue leak inspection; initiate SW loop separation if necessary; initiate controlled shutdown if pressure not stabilized
40 psig	trip the reactor, proceed to E-0

The basis of the setpoints are to establish settings corresponding to system leak rates of approx. 1000 gpm and greater. The various pressure settings listed above have been correlated to the expected system flowrate and leakrate for both 2 and 3 pump operation.

[illegible]

ATTACHMENT A

Question No. 17

Question:
"Confirm that, for various modes of SWS operation, with valves 4760 and 4669 in the open position (diesel generator service water cross-connect valves), including both split and cross-connected header configuration, that a passive failure will not jeopardize operability of both diesel generators. Describe the specific SWS configurations that were considered in this regard."

Response:
The failures of low energy piping such as service water or a failure of a check valve to move to its proper location, when called upon to perform its safety function, is considered a passive failure, and therefore does not need to be analyzed. (This is addressed in our 4/6/92 response, under action Item 2). This is also discussed in our cover letter.

ATTACHMENT A

Question No. 18

Question:
"Given the system constraints and limitations that must be satisfied during an event with only one service water pump available, identify any changes that should be made to the existing Technical Specification LCOs and surveillance requirements. For example, the current Technical Specifications only require one loop header to be operable and does not specifically require that a service water pump be operable in each loop header, which is not consistent with the FSAR description that credits both loop headers as being available for redundancy of cooling capability. Also, requirements for split vs. cross-connected loop header operation are not stipulated."

Response:
As previously demonstrated, the service water system is not constrained if one service water pump is operable, assuming the other pump to be the single failure. The current Technical Specifications (3.3.4) require one loop header and two service water pumps with one pump powered from each electrical train. The UFSAR describes the system whereby there exists a service water loop which is split into two trains. The trains are cross-connected between the emergency diesel generators and the CRFCs as described in UFSAR 9.2.1.3. Because of this configuration flow is balanced to the CRFCs and no single active failure could result in loss of service water flow to redundant critical loads. Owing to the cross-connected configuration, the system could be described as having one service water loop with branch lines cross-connected so that either of the two individual 20" supply lines can deliver flow to the redundant critical loads in the event of a single active failure. Thus, under single active failure requirements to which Ginna was licensed, the other SW train could supply the necessary heat loads following the limiting single failure. It is noted that the single failure of low energy piping such as service water was not required to be assumed in the licensing of Ginna. RG&E maintains the current Technical Specifications are correct as approved.

ATTACHMENT A

Question No. 19

Question:
"Supplemental response to GL 89-13 dated June 1, 1992, only lists one SFP heat exchanger as "critical". This is not consistent with the UFSAR description of critical heat loads which lists two heat exchangers. Explain."

Response:
RG&E acknowledges this inconsistency. As a matter of background, SWSOPI unresolved item 91-201-04 was written relative to the safety classification of spent fuel heat exchanger A. RG&E provided clarification in Attachment 1 of our letter response to 91-201-04 dated April 6, 1992. The NRC closed this unresolved item in Inspection Report 50-244/92-12 dated October 13, 1992. Thermal performance testing of spent fuel pool heat exchanger B is included in the GL 89-13 program as requested by item II of GL 89-13, since it provides a safety related heat removal function. The use of the term "critical in the UFSAR was retained, because that term was used in the NRC SER for SEP Topic IX-3, Station Service and Cooling Water Systems, dated November 3, 1981. A footnote on page 8 of that SER identifies that "critical" refers to a heat load that the licensee has designated as safety-related. That SER was written prior to our installation of the new safety-related spent fuel pool cooling loop (loop B). The existing loop A together with a skid mounted unit comprise a backup to the safety related loop; the loop A heat exchanger does not provide a safety related heat removal function and is not included in the GL 89-13 test program.

Since the UFSAR Table 9.2-2 contains a footnote "a" next to the "spent fuel pool heat exchangers (2)" service water load, implying that both heat exchangers are critical, the UFSAR table will be corrected to remove this inconsistency to specify only the B heat exchanger as critical.

ATTACHMENT A

Question No. 20

Question:
"Identify specifically what service water flow rates are required to be established through each heat exchanger being tested during heat exchanger performance testing that is periodically performed. Also, state the worst-case fouling factors that are assumed for each of these heat exchangers and the basis for the values being credited."

Response:
Below are the Service Water System heat exchangers currently included in the thermal performance testing program with the service water flow rates required during testing:

Heat Exchanger	Required Service Water Flow Rates
Component Cooling Water Heat Exchangers	Service water flow is throttled to achieve $\Delta T_{SW} \geq 10°F$ (typically 2500 gpm - 4000 gpm).
Spent Fuel Pool Heat Exchanger "B"	1600 gpm (design) plus service water flow is throttled to achieve $\Delta T_{SW} \geq 10°F$ (typically 600 gpm).
Diesel Generator Coolers (Jacket Water and Lube Oil)	Coolers are tested with service water flow at 350 gpm - 400 gpm.
Containment Recirculating Fan Coolers (Including Motor Cooler Coils)	Coolers are tested with service water flow at 500 gpm plus at 250 gpm - 500 gpm (typically 300 gpm) to ensure $\Delta T_{SW} \geq 10°F$)
Standby Auxiliary Feedwater Pump Room Cooling Units	Units are tested with service water flow at 25 gpm (design) plus at 10 gpm - 15 gpm.

Based on thermal performance test data, actual heat exchanger fouling factors are determined and compared to the design values. As-tested fouling factors below the design values indicate the heat exchanger is capable of performing its intended function. As-tested fouling factors above the design values indicate corrective action is necessary. This testing methodology does not require fouling factor assumptions as it directly determines heat exchanger condition and incorporates test instrument inaccuracies.

[illegible]