NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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TO: A. SCHWENCER

FROM: ROCHESTER GAS & ELEC. CORP.
ROCHESTER, N.Y.
L.D.WHITE,JR.

DATE OF DOCUMENT: 2/24/77

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(2P)

ENCLOSURE

RESPONSE TO STAFF POSITIONS AND ADDITIONAL INFORMATION REQUESTS REGARDING THE REACTOR VESSEL OVERPRESSURIZATION.....

(27P)
(1 SIGNED CY. RECEIVED)

ACKNOWLEDGED

PLANT NAME: RE GINNA # 1

FOR ACTION/INFORMATION

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NRC FORM 195 (2-76)

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

LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

February 24, 1977

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

In your letter of January 10, 1977 you directed us to select one of two options for implementation of hardware improvements to meet your objective of improved overpressure protection in all operating PWR facilities by the end of 1977. These hardware improvements are to be in addition to the procedural and administrative measures which we have already instituted and which you concluded will help prevent any future pressure transients.

We are designing and will procure and install components for an overpressure protection system which will meet the design criteria outlined in Attachment 2 to your January 10, 1977 letter, to the extent practical given the existing equipment configuration of R. E. Ginna. The system will be based on the assumption that a single pressurizer power operated relief valve will mitigate the consequences of overpressure transients not caused by inadvertently discharging an accumulator. To the extent practical, we will install redundant overpressure protection components which are seismically qualified, which meet IEEE-279 criteria and which can be tested on a schedule consistent with the frequency of use for overpressure protection. The objective will be to provide a system which is not vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

No shutdown has been scheduled for Ginna during the coming year other than the 1977 Spring refueling outage which is to commence in April. Components for the overpressure protection system are not available for that shutdown. Because the overpressure protection system will not be used while the plant is operating and is of value only during shutdowns, we do not plan to schedule a special outage to install this equipment.

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Therefore we plan to install the long term hardware improvements described above during the first scheduled shutdown after December 31, 1977, provided the system components have been delivered to us and you have concurred with our response to your letter of February 14, 1977. We will work expeditiously to have the components available by the end of this year. Should an unscheduled outage of sufficient duration to install the system occur before that shutdown but after the system components have been delivered, we will install the system at that time.

Enclosed is the additional information which you requested in Attachment 1 to your January 10, 1977 letter.

Sincerely yours,

L. D. White, Jr.

L. D. White, Jr.

RESPONSE TO

STAFF POSITIONS AND ADDITIONAL INFORMATION REQUESTS
(Attachment 1 to A. Schwencer letter dated January 10, 1977)

REACTOR VESSEL OVERPRESSURIZATION

R. E. GINNA UNIT NO. 1
DOCKET NO. 50-244

February 24, 1977

R. E. GINNA
NUCLEAR POWER PLANT

QUESTION 1: The staff considers it essential that all plant operators (i.e., reactor operators, equipment operators, Instrument & Control personnel) be made aware of the details of the pressure transients which have taken place at all PWR facilities. POSITION: Formal discussions should be held with the operator to review the causes of past pressure transients that have occurred at other operating PWR facilities. Your discussions should include the plant conditions at the time, the mitigating action that could have been or was taken, and the preventive measures that could have been taken to avoid the event and the steps taken to prevent similar, further occurrences. Plant similarities and distinctions should be identified along with how these relate to plant start-up, shutdown, and testing operations. With regard to this position, you are requested to provide the following information:

a. If you have not already completed the required formal discussion, when will you do so?

b. How will the discussions be held?

c. Of the past PWR Appendix G violations that have occurred at PWR facilities and which are described in License Event Reports, identify which are not credible in your plant due to equipment differences. Provide a description of the distinctions.

d. Describe, in detail, how you are reducing the likelihood of the other remaining credible events. Furnish schematics, diagrams or procedural summaries necessary to support the effectiveness and reliability of these measures.

RESPONSE: 1.a.
The formal discussions called for in the Staff Position were begun the week of February 8, 1977 and are scheduled to be completed the week of March 8, 1977.

RESPONSE: (Continued)

1.b.
The discussions are being held during Operator Requalification Training and will include all licensed operating personnel and Instrumentation and Control technicians. The discussions include similarities to and differences from the R. E. Ginna Plant and emphasize the actions that could have been or were taken to reduce the impact of the overpressurization event. Preventive measures which can be taken to prevent similar events are also discussed. Plant startup, shutdown, and testing conditions are related to the potential for overpressurization events.

1.c.
Of the past PWR Appendix G violations that have occurred at PWR facilities, six are considered to be incredible events at R. E. Ginna because of equipment differences.

Events similar to those which occurred at Beaver Valley Unit No. 1 on February 24, 1976, Turkey Point Unit No. 3 on December 3, 1974, and Zion Unit No. 2 on September 18, 1975 were caused by automatic isolation of the RHR system. There is no automatic isolation of RHR at Ginna.

The event at Indian Point Unit No. 2 on May 18, 1973 resulting from closure of certain air operated valves in the reactor coolant letdown system was caused by freezing of moisture in the air supply line. At Ginna the entire instrument air system is inside heated buildings and the air passes through air dryers before being piped to the point of use.

The event at Trojan on July 22, 1975 resulted from the RHR suction valve from the reactor coolant system being closed and isolating system letdown while the positive displacement charging pump was operating. At Ginna, isolating the RHR suction line from the reactor coolant system would not isolate letdown. A flow path exists from the reactor coolant system through the RHR discharge line to the RHR relief valve. This relief valve will limit system pressure to 600 psig and prevent exceeding technical specification limits whenever the reactor coolant system temperature is greater than 175°F.

RESPONSE: (Continued)

The event at Peach Bottom Unit No. 2, a boiling water reactor, on March 6, 1974 is considered not to be applicable to Ginna, a pressurized water reactor, because of the different reactor types and operating equipment. It is possible for an operator to cool the primary system while holding a constant system pressure although an alarm has been installed to alert the operator if he approaches Technical Specification limits.

1.d.
The remaining events which remain credible, although improbable, at Ginna have been caused by inadvertent letdown isolation, reactor coolant pump starts, inadvertent accumulator injection, safety injection tests or improper procedures or personnel error.

The likelihood of letdown isolation overpressurization events similar to those which occurred at Indian Point Unit No. 2 on 2/17/72 and 4/6/72, Prairie Island Unit No. 2 on 11/27/74, St. Lucie Unit No. 1 on 8/12/75, D. C. Cook on 4/14/76 and Beaver Valley Unit No. 1 on 3/6/76 is reduced by providing a second letdown path. The cooldown procedures require that the RHR valves to the primary system be opened while there is a steam bubble in the pressurizer. The procedures also require that the RHR system be connected to the letdown system which contains a 600 psig relief valve. The analysis in Appendix A to this attachment has shown that this valve and the piping connecting it to the RHR system are capable of relieving the full output of two charging pumps (120 gpm) over the design temperature range of the RHR system. This valve and piping can relieve the output of three charging pumps (180 gpm) when the system temperature is below 200°F. Plant procedures control the use of the charging pumps within these guidelines while the RHR system is in operation.

The plant heatup procedures call for a steam bubble to be formed in the pressurizer prior to isolation of the RHR system. There is no automatic function which will isolate the RHR system upon rising primary system pressure. Thus the relief valve will remain available.

The likelihood of overpressurization events caused by reactor coolant pump starts similar to those at Indian Point Unit No. 2 on 3/8/72 and 1/23/74, Prairie Island Unit No. 1 on 10/31/73 and St. Lucie Unit No. 1 on 6/17/76 has been reduced by procedural control of the

RESPONSE: (Continued)

reactor coolant pumps. If all the reactor coolant pumps have been idle for more than 5 minutes and the reactor coolant temperature is greater than the charging and seal injection water temperature, a steam bubble must be formed in the pressurizer before starting a reactor coolant pump. In addition, if the reactor coolant is being cooled by the RHR system and all the reactor coolant pumps have been idle so that a nonuniform temperature distribution may have occurred in the reactor coolant system, a steam bubble must be formed in the pressurizer prior to starting a reactor coolant pump. Our cooldown procedures call for the last reactor coolant pump to be stopped only after the reactor coolant system has reached approximately 150°F, thus minimizing any temperature differentials which could occur between system components. On startup, the operators normally throttle back RHR cooling and allow the reactor temperature to rise.

Overpressurization events similar to those at Indian Point Unit No. 2 on 2/22/74 and Prairie Island Unit No. 1 on 1/16/74 resulting from inadvertent discharge of an accumulator have been reduced in likelihood by deenergizing the accumulator isolation valve at all times when the reactor coolant system pressure is less than approximately 1000 psig. The cooldown procedures require that the discharge valves between the accumulators and the primary system be closed at less than 1800 psig and that power be removed from these valves at approximately 1500 psig. The heatup procedures do not allow the accumulator isolation valves to be reenergized and opened until primary system pressure is greater than 1000 psig.

An overpressurization event caused by an improperly aligned safety injection system during tests similar to that at Point Beach Unit No. 2 on 12/10/74 is unlikely at Ginna because of the procedural controls applied to periodic tests. The test procedures require that whenever reactor coolant system pressure is less than 1600 psig during safety injection pump tests, the pump discharge valves must be closed and the procedure initialed by the operator to signify that the valves have been properly aligned.

RESPONSE: (Continued) The remaining six overpressurization events whose causes are known resulted from what appear to have been poor operating procedures, operator improvised procedures or operator error. Ginna operations and tests procedures are reviewed by the Operations Engineer or Tests and Results Engineer respectively and are reviewed by the Plant Operations Review Committee. Operators are not allowed to improvise or deviate from established procedures. Procedures may be revised on a temporary basis only as detailed in Technical Specification 6.8.3. Operators are keenly aware of overpressurization events because of recent increased emphasis and training.

The effectiveness and reliability of the measures which have been taken to reduce the likelihood of overpressurization events is demonstrated by the Ginna operating history.

R. E. GINNA
NUCLEAR POWER PLANT

QUESTION 2: The majority of the reported pressure events have occurred while the plants were operating in a water solid condition. POSITION: The staff will require that operations during which the plant is maintained in a water solid condition be minimized or if possible eliminated. Those operations in which the plant is in a water solid condition must be fully justified.

Attachment 1 to your October 15 letter and your October 26, 1976 letter to the NRC discuss various requirements that minimize the operation of R. E. Ginna while in a solid-water condition. Based on the staff's position stated above, and the review of your letters, the following additional information is requested.

a. Your justification for not partially draining and venting the pressure during cold shutdowns of less than seven days duration.

b. Describe the procedures, evolutions or situations that require the plant be maintained in a water solid condition. Also provide reasons why a nitrogen, air or steam bubble cannot be maintained in these situations.

c. Include sufficient background or supplementary information such as system diagrams, procedure summaries and descriptions of equipment operation to justify your need for operating the plant water solid.

RESPONSE: 2.a.
Draining and venting the pressurizer lengthens the amount of time required to return the plant to operation by approximately 8 hours. The additional time is required to "burp" the pressurizer and to return the water chemistry to the specified limits. Cold shutdowns of short duration are occasionally required to repair or adjust equipment. Occasionally unplanned outages occur which may require investigation to determine the extent of the difficulties. Seven days is sufficient time to locate and correct minor problems or to determine that draining and venting the primary system is required. Draining and venting the primary system generates additional liquids and gasses which must be processed. The gasses are eventually discharged. Thus, a limit of seven days will minimize the length of time for water solid plant operation and also minimize the quantity of radioactive effluent.

RESPONSE: (Continued)

2.b.

The plant must be operated in a water solid condition during filling and venting of the reactor coolant system, hydrostatic pressure testing of the reactor coolant system boundary and during plant heatup including reactor coolant pump operation prior to bringing the reactor coolant system within the chemistry specifications.

Nitrogen, air or a steam bubble is not maintained in the pressurizer during these conditions for the following reasons. During filling and venting the pressurizer vent is used to release trapped air from the system. During the hydro test visual inspections for leaks are required on valves and piping on top of the pressurizer. Technical Specifications prohibit drawing a bubble during plant heatup until reactor coolant system chemistry specifications have been met. Air or nitrogen in the pressurizer at higher temperatures is prohibited and therefore must be excluded while bringing the coolant chemistry within limits.

2.c.

The need for water solid operation during filling and venting is evident.

The reason for hydrostatically testing with water solid conditions is clear, however, it should be noted that this procedure is carried out at temperatures high enough so that the pressurizer safety valves protect the system from exceeding Technical Specification limits.

Reactor coolant pump operation while water solid is required to circulate the coolant to aid in meeting the chemistry specifications. As noted in the response to question 1, however, our cooldown procedure calls for a reactor coolant pump to be run until the system temperature reaches approximately 150°F. This minimizes any temperature differences which may exist between the reactor and steam generators and minimizes the likelihood of an overpressurization event.

QUESTION 3: The inadvertent operation of SIS components during cold, shutdown conditons has been responsible for a major portion of the overpressure incidents. POSITION: Based on the licensee submittals, the recent November 3-5, 1976 meetings, and discussions with NSSS vendors, the staff will require the de-energizing of SIS pumps and closure of SI header/discharge valves during cold shutdown operations. Those situations during which this position cannot be met must be described and be fully justified.

Your letters to the NRC dated October 15 and 26, 1976, discuss the relignments you make to the SIS during plant cooldowns and heatups. The staff has reviewed these actions, and requests the following additional information:

a. A schematic diagram of the SIS showing the flow-paths into the RCS.

b. The head-flow characteristics of each of the SIS pumps.

c. Justification for placing the high head SI pump control switch in "pull-stop" and not de-energizing the pump.

d. Identify on the schematic diagrams the pumps and the valves to be closed, de-energized, or disabled, (e.g., "pull-stop").

e. Your time schedule for implementing the procedural or administrative changes that require the alignment shown on the above diagram.

f. Indicate all circumstances for which these SIS pumps and valves may not be isolated, de-energized and/or disabled. For those situations, describe the manner in which SIS injection would be prevented.

g. The location of the high head SI pump control switch, and all other places from which the pump may be controlled (e.g., Motor control center, locally, etc.).

QUESTION 3: (Continued)

h. The location of all remaining SIS component power supply breakers, and the places from which they can be controlled.

i. Describe the position indication and status signals which would be lost if the components were (1) de-energized or (2) disabled by placing control switch in "pull-stop".

j. Describe in detail, the administrative procedures which will be used to assure proper equipment alignment and the supervisory personnel responsible for maintaining control.

k. Provide the reasons for not closing and removing power from the accumulator MOVs at the same RCS pressure (as noted in your October 26 submittal).

l. Describe the impact on overall plant operations if you routinely lowered accumulator nitrogen pressure when in a cold shutdown condition.

RESPONSE:

3.a
A schematic of the safety injection system is given in Figure 6.2-1 of the R. E. Ginna FSAR.

3.b.
The head-flow characteristics of the safety injection pumps are given in Figure 6.2-13 of the R. E. Ginna FSAR.

3.c.
Placing these pumps in "pull-stop" will prevent inadvertent actuation of the pumps and yet leave them available from the control room should they be needed. This method of operation provides greater core protection and assurance of safe shutdown than does pulling the pump breakers.

3.d.
During plant cooldown from hot shutdown to cold shutdown the accumulator discharge valves, 865 and 841, are closed and the breakers are pulled. Safety injection pump discharge valves to the cold legs, 878B and 878D, are closed and the safety injection pumps are put in the "pull-stop" position. Technical Specifications require that safety injection pump discharge valves to the hot legs, 878A and 878C, be closed and deenergized for plant operation. They are not repositioned during cooldown.

RESPONSE: (Continued)

3.e.
The alignments described in the response to question 3.d are those required by existing procedures.

3.f.
A high head safety injection pump is used during cold shutdown to fill the accumulators to their normal operating level. The procedure for filling the accumulators directs that the pump discharge valves 878A, 878B, 878C, and 878D be closed prior to starting a pump.

A refueling shutdown surveillance procedure covering diesel generator loading is conducted with the safety injection pumps energized and running in the recirculation mode. Prior to the test the 878A, 878B, 878C and 878D valves are closed or verified closed and the D.C. control circuit fuses pulled. The A.C. power is also removed from the valves.

3.g.
The high head safety injection pump control switches are located in the control room. There are no local control switches outside the control room, however, the pump breakers at the 480 volt busses may be closed manually with the aid of a special handle.

3.h.
The power supply breakers for the valves discussed in 3.d above are located in the auxiliary building in motor control centers 1C and 1D. They are controlled only from the main control board.

3.i.
Valves that are deenergized by pulling breakers retain all of their normal status light indication. This status indication is provided by the D.C. control circuitry. Light indication is lost only when the D.C. control power fuses are removed at the motor control center breaker panel.

When the safety injection pumps are placed in "pull-stop" the switch indicating light goes out and a "safeguard equipment locked off" annunciator is alarmed.

RESPONSE: (Continued)

3.j.
Administrative procedure A-30.4, Plant Procedure Adherence Requirements, governs the use of the system alignment procedures. Deviations from established procedures are permitted only by the approval authority outlined in Technical Specification 6.8.3. Proper valve alignment is documented by the completed procedures. The supervisory personnel responsible for maintaining control are the Shift Foreman, Operations Supervisor, and Operations Engineer.

3.k.
There is no reason except for procedure flow and continuity. The valves are closed and power removed at pressures above the discharge pressures of the accumulators.

3.l.
The impact on overall plant operations of routinely lowering the accumulator nitrogen pressure to one atmosphere during cold shutdown would be the loss of approximately two days to refill the accumulators to operating pressure. This operation would cost several thousand dollars not including purchase of replacement power during those times when it would be a critical path operation. The operation involves an additional operator full time and a fitter part time to restore the accumulator nitrogen pressure.

QUESTION 4: The staff has noted that several Appendix G violations have occurred during component or system tests while in cold and shutdown conditions. In this regard, please address the following questions.

a. What components or systems that could cause overpressure transients, are routinely tested while in a cold shutdown condition?

b. What extra measures are taken to prevent an overpressure event during these tests?

RESPONSE: 4.a.

The Diesel Generator Load and Safeguard Sequence Test is conducted at cold shutdown. During this test the high head safety injection pumps are operated on recirculation flow. The safety injection pump discharge headers are pressurized to approximately 1500 psig up to the hot and cold leg injection valves 878A, 878B, 878C and 878D.

4.b.

To prevent an overpressurization event during this test the hot and cold leg injection valves are procedurally maintained closed with both D.C. control and A.C. motive power off.

QUESTION 5: The staff believes that a high pressure alarm used during low RCS temperature operations is an effective means to attract the operators's attention to a transient in progress. POSITION: The staff is requiring that if it is not presently installed, it must be as soon as possible.

Based on the staff's position stated above, and the review of your October 15 and 26, 1976 letters, the items listed below should be addressed:

a. The alarm setpoint, mode of annunciation and sensors used.

b. A synopsis of the system modifications that were necessary to furnish the alarm.

c. Your means to assure the alarm's availability during all water-solid operations, and to minimize its downtime for all other cold shutdown conditions.

RESPONSE: 5.a.
Two setpoints are incorporated into the alarm. One is variable and follows the Technical Specification limit. The other alarms at a given differential pressure, determined by the operator, below the Technical Specification limit. Both setpoints alarm the bell and light on the computer typewriter. The sensors used to generate the alarms and the Technical Specification setpoint are the primary coolant loop wide range pressure transmitter and two cold leg temperature sensors.

5.b.
The alarm was installed by the addition of computer software. No hardware changes were required.

5.c.
It is normal plant practice not to perform computer maintenance during changes of plant operating modes. Computer maintenance is normally performed when the plant is not in a water solid conditon.

QUESTION 6: The RHR (or SCS) is normally connected to the RCS and operating when the plant is in a cold shutdown condition. The inadvertent isolation of the RHR system while water solid has caused a number of overpressure transients, and the RHR safety valve has actually terminated others. The RHR (or SCS) therefore plays an important part in the initiation and possible mitigation of the PWR overpressurizations. Accordingly, we request the following additional information:

a. RHR (or SCS) design pressure.

b. A description of the system isolation valves and their arrangement (e.g., number and configuration of valves installed, and pneumatic or motor operated).

c. Interlocks, interlock setpoints, and alarms associated with each isolation valve.

d. Nominal stroke time of isolation valves.

e. The setpoint and capacity of RHR (or SCS) relief and safety valves.

f. All pressure alarms, setpoints and associated annunciation for the system.

RESPONSE: 6.a.
The RHR system design pressure is 600 psig.

6.b.
The RHR suction and discharge valves connecting this system to the primary coolant system are shown on Figure 9.3-1 of the R. E. Ginna FSAR. The reactor coolant system suction supply to the RHR pumps is from the hot leg of loop A through motor operated valves MOV 700 and MOV 701 in series. The RHR pump discharge return to the loop B cold leg of the reactor coolant system is through two series motor operated valves, MOV 720 and MOV 721.

RESPONSE: (Continued)

6.c.
Permissive interlocks required to open the four RHR system isolation valves are listed below.

MOV 700	(1) Reactor coolant system pressure must be less than 410 psig
	(2) RHR suction valves MOV 850 A and MOV 850 B from the containment sump must be closed.
MOV 701	(1) RHR suction valves MOV 850 A and MOV 850 B from the containment sump must be closed
	(2) the valve is operated by a key switch
MOV 720	No interlocks exist but the valve is operated by a key switch
MOV 721	(1) Reactor coolant system pressure must be less than 410 psig

No interlocks are associated with valve closure. There are no automatic functions which close the valves and no alarms generated by the valves.

6.d.
The nominal stroke time of the RHR isolation valves is 3 minutes.

6.e.
The setpoint of the RHR relief valve is 600 psig. The design capacity is 70,000 lbs/hour. A more detailed analysis of the relief valve is presented in Appendix A.

6.f.
The pressure alarms, setpoints and associated annunciation of the RHR system are a high pressure alarm at 550 psig and a reactor coolant system interlock pressure alarm at 410 psig.

QUESTION 7: Reactor coolant system heatups, resulting from improper operation of the reactor coolant pump (RCP) while in a cold, shutdown and water solid condition, have been responsible for approximately 15% of the RCS overpressurization events. POSITION: We will require that all licensees include adequate provisions to prevent RCP starts while in a water solid condition unless absolutely necessary. In those cases where the RCP starts cannot be avoided, the licensee should take appropriate steps to determine and minimize the RCP temperature profile.

Based on the position stated above, please provide the following information:

a. Describe the normal operating conditions during which the RCS is maintained water solid with all RCP's stopped (e.g., fill and vent, pressurizer cooldown).

b. For each of the above procedures, justify your inability to establish a N_2, air or steam bubble in the pressurizer prior to the start of the first RCP.

c. What are the limits associated with system temperatures before the first RCP can be started in a solid RCS?

d. Specify the instruments utilized to determine the RCS temperature profile.

e. Provide the necessary schematics and procedural descriptions that show what your actions would be to bring the RCS to an isothermal condition.

f. Summarize any other measures you take to reduce possible RCS pressure spikes during RCP starts, (e.g., open all letdown orifice isolation valves, stop makeup flow, etc.)

RESPONSE: 7.a.
The reactor coolant system is maintained water solid with all reactor coolant pumps stopped during (1) normal cold shutdown below approximately 150°F when the reactor coolant system is not to be opened, (2) fill and vent operations and (3) the start of system heatup.

RESPONSE: (Continued)

7.b.
A steam bubble can be formed in the pressurizer prior to the start of a reactor coolant pump provided that the reactor coolant chemistry is within technical specification limits. If the chemistry is not within the required limits a reactor coolant pump must be started prior to drawing a pressurizer bubble in order to mix the coolant and the chemicals which are added to restore proper system chemistry. Air or N_2 bubbles are not used because of the problems they cause in meeting chemistry requirements. Thus, following those system operations listed above in 7.a., a steam bubble will be formed in the pressurizer before starting a reactor coolant pump provided that the reactor coolant system chemistry meets Technical Specification limits.

7.c.
There are no formal limits on reactor coolant system temperatures prior to the start of the first reactor coolant pump in a water solid system. However, the operating methods discussed in 1.d. minimize temperature differentials between system components.

7.d.
Instruments used to determine the reactor coolant system temperature profile include pump seal injection temperature, charging inlet flow temperature, hot and cold leg temperatures and RHR heat exchanger inlet and outlet temperatures.

7.e.
As discussed in 1.d., the reactor coolant system and the steam generators are cooled to approximately 150°F before the last coolant pump is stopped. The reactor temperature is normally allowed to rise prior to restarting the first coolant pump.

If the charging and seal injection water temperature is less than the reactor coolant temperature and both reactor coolant pumps have been stopped for more than five minutes, a steam bubble is drawn in the pressurizer before starting the first reactor coolant pump. This method obviates the need for an isothermal condition.

7.f
Before starting a reactor coolant pump in a water solid condition, the normal letdown isolation valve and all 3 orifice isolation valves are opened. The operator takes manual control of the letdown pressure control valve. The procedures also caution the operator that it will aid in reducing pressure spikes to reduce charging pump speed and stop the running RHR pump.

APPENDIX A

As can be seen in Figure 9.3-1 of the FSAR, the RHRS is connected to the loop A hot leg on the suction side and the loop B cold leg on the discharge side. The design pressure and temperature of the RHRS is 600 psig and 400°F. The design basis with regard to overpressure protection for Ginna Station's RHRS is to prevent opening of the RHRS isolation valves when Reactor Coolant System (RCS) pressure exceeds 450 psig and to provide relief capacity sufficient to accommodate thermal expansion of water in the RHRS and/or leakage past the system isolation valves.

Westinghouse Electric Corporation has performed an analysis of incidents which might lead to overpressurization of the RHRS. Three events were considered in the analysis:

(1) The letdown line is isolated from the RHRS solid, the RHRS is in operation, and the charging pumps are running.

(2) During cooldown utilizing the RHRS, one cooling train suffers a failure at a time when the heat generation rate exceeds the heat removal capability of the remaining cooling train.

(3) Pressurizer heaters are energized while the RHRS is in operation.

The results of the analysis are presented below.

Event No. 1

The first event considered in the RHRS relief analysis is the isolation of letdown flow with charging pumps running and with the plant in a solid water condition and the RHRS in operation. For this analysis it was assumed that, once open, RV203 shuts at 5% blowdown, corresponding to a pressure at 570 psig for a setpoint of 600 psig. Flow through RV203 at this pressure would be approximately 185 gpm with the system below 200°F and 130 gpm with the system at 380°F, the valve design point, based on letdown temperature.

The analysis shows the combined capacity of the 3/4 inch and 2 inch branches connecting the RHRS to CVCS (see Figure 1) and relieving through RV203 to be sufficient to pass the full output of two charging pumps (120 gpm) over the design temperature range of the RHRS (see Figure 2). Sufficient capacity to relieve the output of three pumps (180 gpm) when the system is below 200°F was also shown. When momentary transient pressures due to relief valve cycling are neglected, the RHRS will remain below the maximum allowable pressure of 660 psig (110% of design pressure) during relief operation. This event can be terminated, of course, by establishing letdown flow.

Plant procedures have been revised to prohibit use of three charging pumps while the RHRS is in operation.

Event No. 2

The second event analyzed is that during a cooldown utilizing the RHRS, one cooling train suffers a failure at a time when the heat generation rate exceeds the heat removal capability of the remaining cooling train.

The assumptions made for this analysis are :

a) RCS temperature and pressure are 350°F and 425 psig prior to the event.

b) The RHRS is brought into service at 2 hours after shutdown (assumes a cooldown rate of 100°F/hr).

c) Immediately after initiating RHRS operation, it is necessary to isolate one RHR train (one RHRS pump and one RHRS heat exchanger).

d) For purpose of analysis, no credit is taken for heat removal via the steam generators once the RHRS is brought into service.

e) One RHRS train (one pump and one heat exchanger) removes heat at its design rate. This is conservative because the high temperature difference 2 hours after shutdown will result in more heat transfer.

f) Heat input to the reactor coolant system consists of decay heat (constant rate at 2 hours after shutdown), thick metal heat, and heat from two running coolant pumps (3 MW each).

g) RHRS relief valve opens at 600 psig and reaches a relief rate of 139.2 gpm at 660 psig.

h) No letdown flow is assumed.

i) The pressurizer is water-solid prior to the event.

The results of this analysis are depicted in Figure 4. This analysis indicates a release of 90.2 gpm through the RHRS relief valve which with heat removal at the design rate of one RHR train (one pump and one heat exchanger) would limit the coolant pressure to 638.8 psig. It is anticipated that the relief valve may chatter during the course of this event. Figure 4 shows that the pressure quickly rises to a plateau and remains there. The constant heat source, described in assumption (f) causes a linear rise in average coolant temperature. This temperature and pressure response indicates that the coolant expansion rate is just matched by the water relief through the RHRS relief valve and the heat removal via the RHRS heat exchanger.

Only a small opening in the relief valve is required to limit pressure under these conditions. If more heat removal by the operational RHRS train is permitted (see assumption e), the relief valve may remain closed.

Event No. 3

The third event analyzed is energizing the pressurizer heaters while the RHRS is in operation. The assumptions made for this analysis are:

a) RCS temperature and pressure are 350°F and 425 psig prior to the event.

b) Only one RHRS train is in service (one RHRS pump and one RHRS heat exchanger).

c) No heat removal via the steam generators.

d) No heat removal via the RHRS.

e) The pressurizer heaters input heat at a constant rate of 800 kw until they automatically shut off when the pressurizer water volume drops below 85 ft^3.

f) A letdown flow of 30 gpm is assumed in order to simulate the conditions under which such an event is likely to occur - heaters are turned on during startup, and letdown flow is permitted in order to draw a steam bubble in the presurizer. The expansion rate due to boiling in the pressurizer is greater than that due to heating the pressurizer.

g) Two reactor coolant pumps are running and contributing 3 MW of heat/pump.

h) RHRS relief valve opens at 600 psig and reaches a relief rate of 139.2 gpm at 660 psig. The relief rate is extrapolated linearly beyond 660 psig.

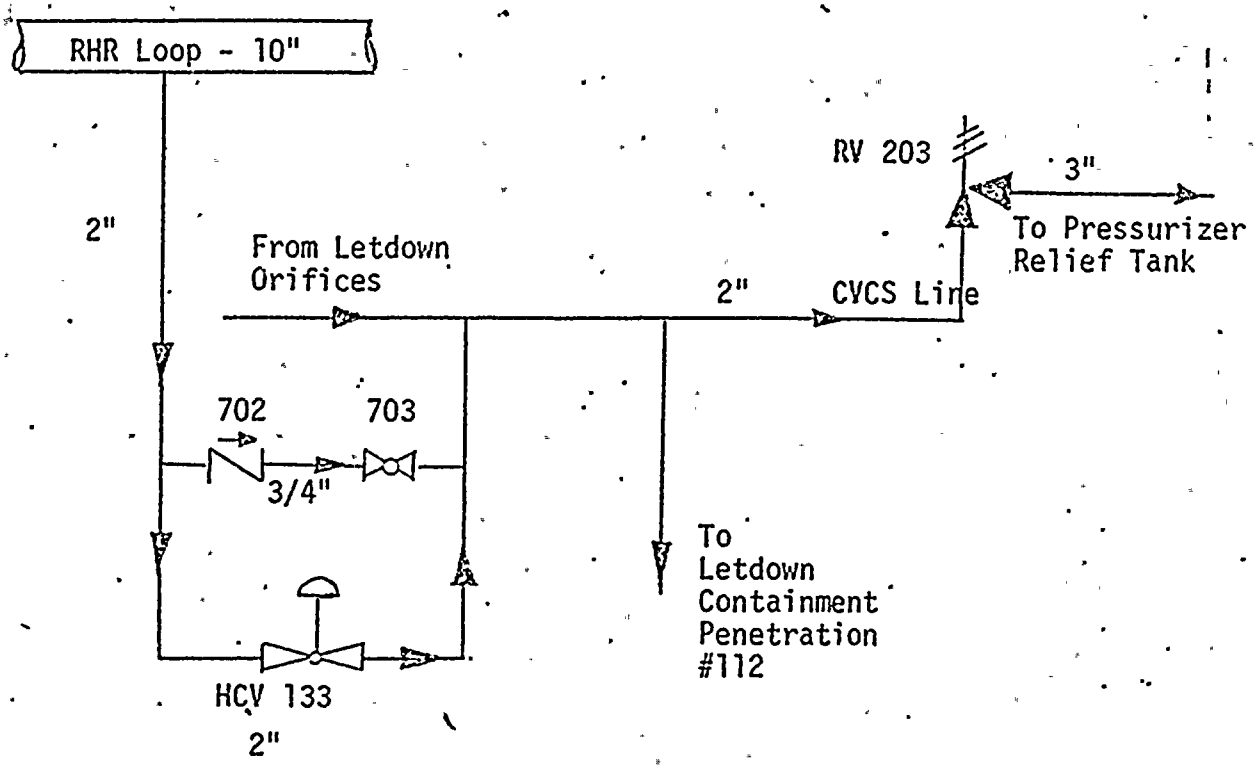
i) The pressurizer is water-solid prior to the event.

j) No decay heat.

k) The operator response to the overpressure alarm on the RHRS, which is set at 550 psi, at ten minutes after the alarm is actuated and turns off the heaters.

As shown in Figure 5, the RHRS pressure remains below 660 psi for a period of time which is sufficiently long so that the operator may reasonably be expected to act. The operator is alerted by the overpressure alarm on the RHRS, by a letdown line hi pressure alarm, and by a reactor coolant loop pressure indicator. The event is terminated once the operator turns off the pressurizer heaters. It is reasonable to assume timely operator action since the pressurizer heaters are in operation during that period of time during RHRS operation of high operator surveillance.

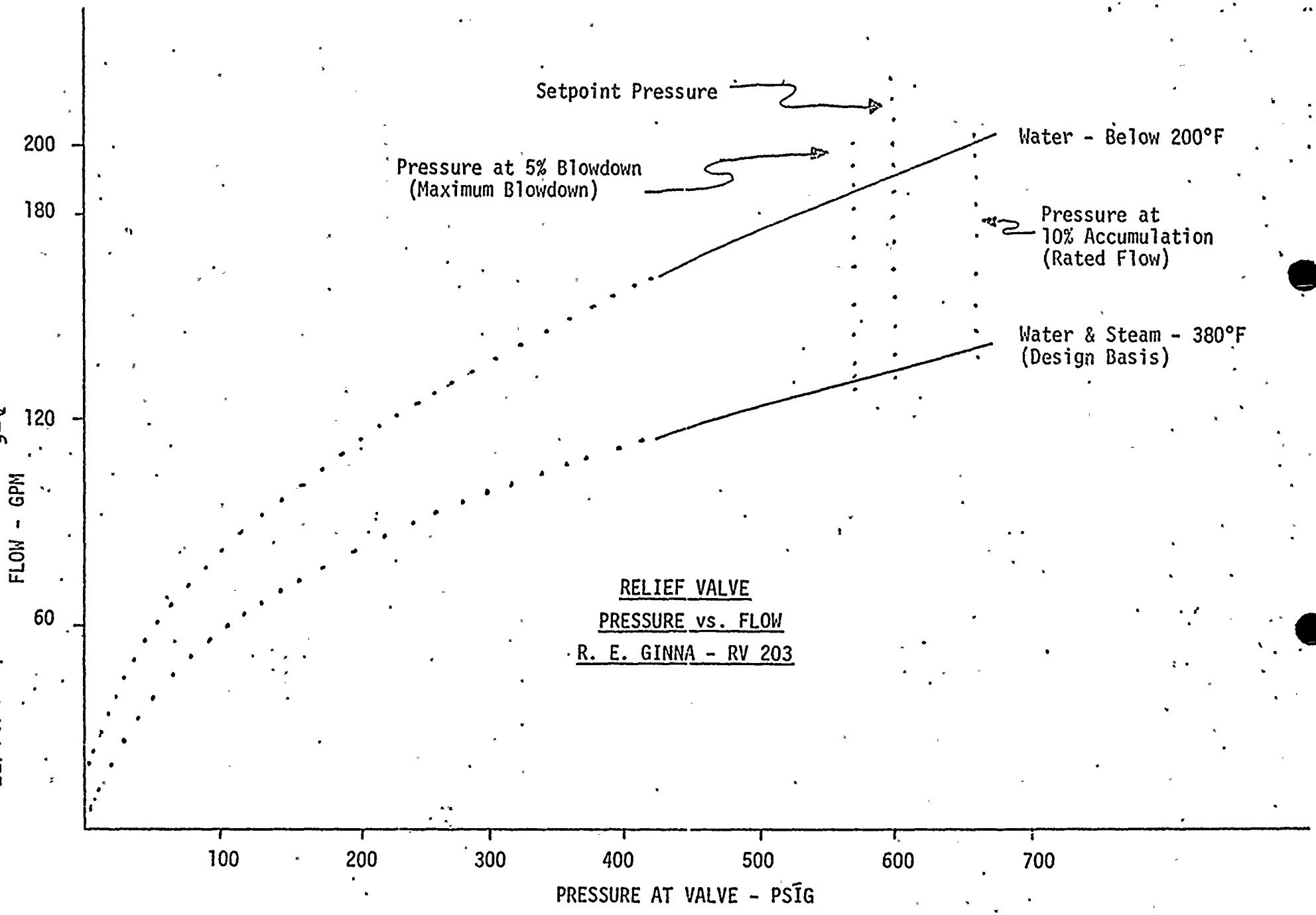
FIGURE 1



CONFIGURATION OF RHRS
RELIEF LINE

2/24/77

FIGURE 2



RELIEF VALVE
PRESSURE vs. FLOW
R. E. GINNA - RV 203

2/24/77

FIGURE 3

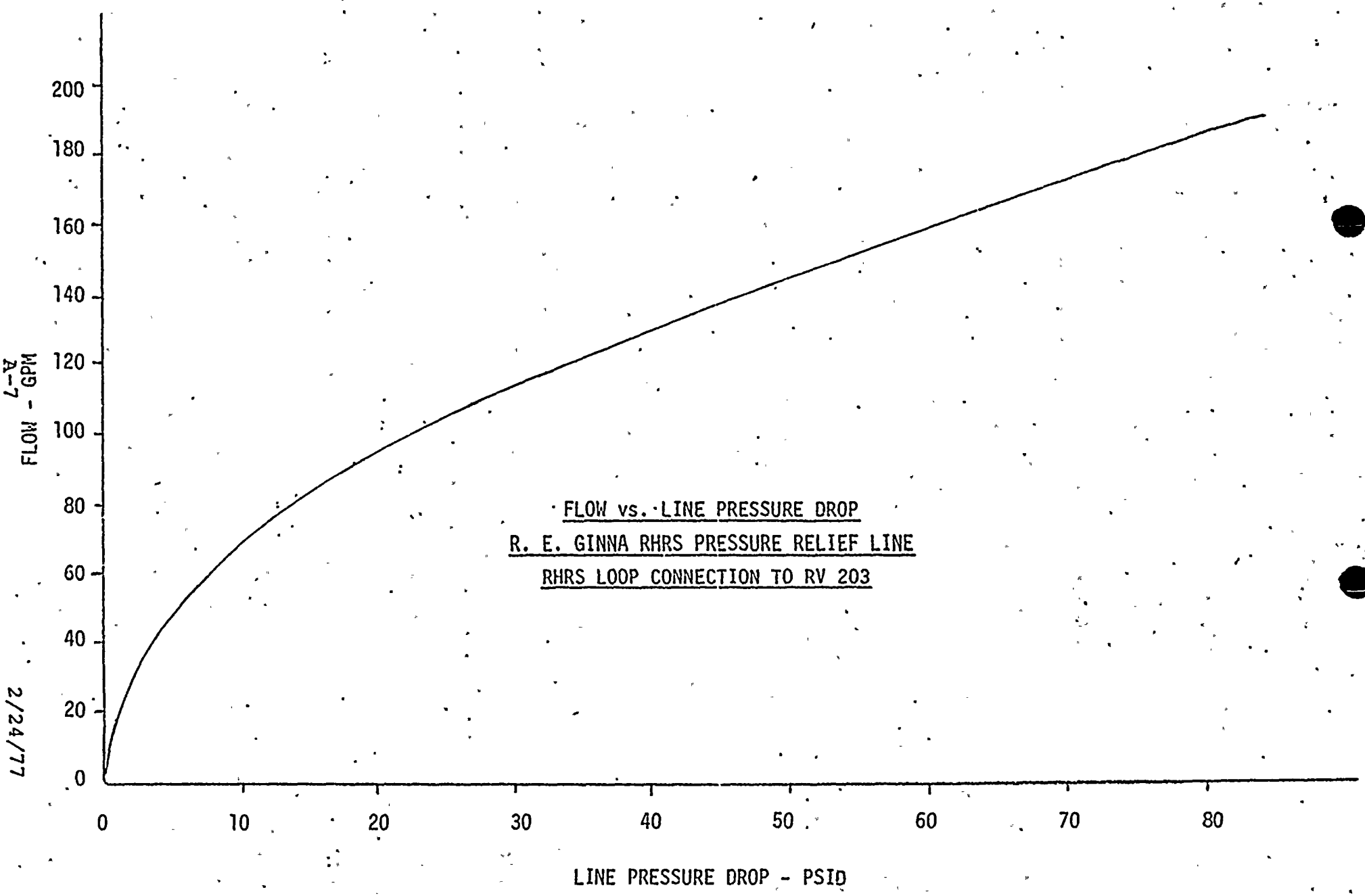


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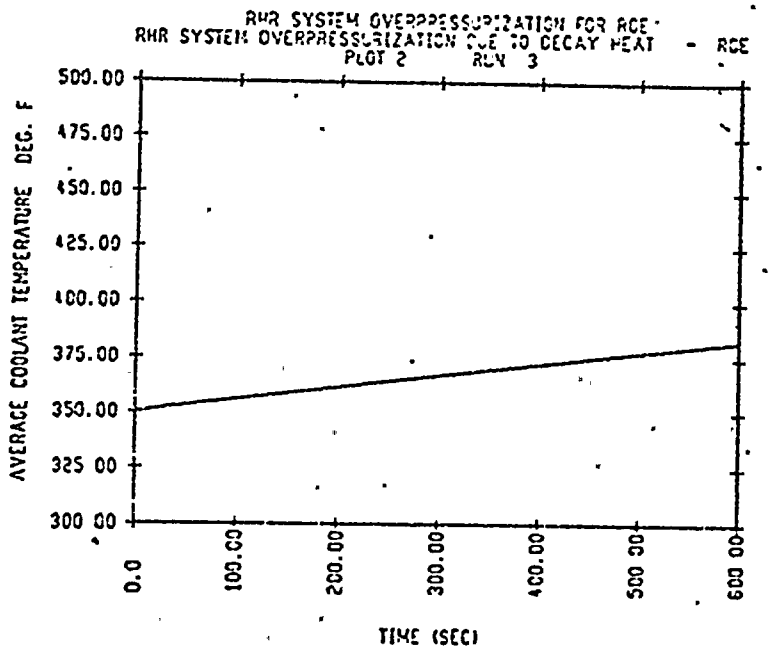
FIGURE 4


RHR SYSTEM OVERPRESSURIZATION FOR RCE
RHR SYSTEM OVERPRESSURIZATION DUE TO DECAY HEAT - RCE
PLOT 2 RUN 3
AVERAGE COOLANT TEMPERATURE DEG. F
500.00
475.00
450.00
425.00
400.00
375.00
350.00
325.00
300.00
0.0
100.00
200.00
300.00
400.00
500.00
600.00
TIME (SEC)

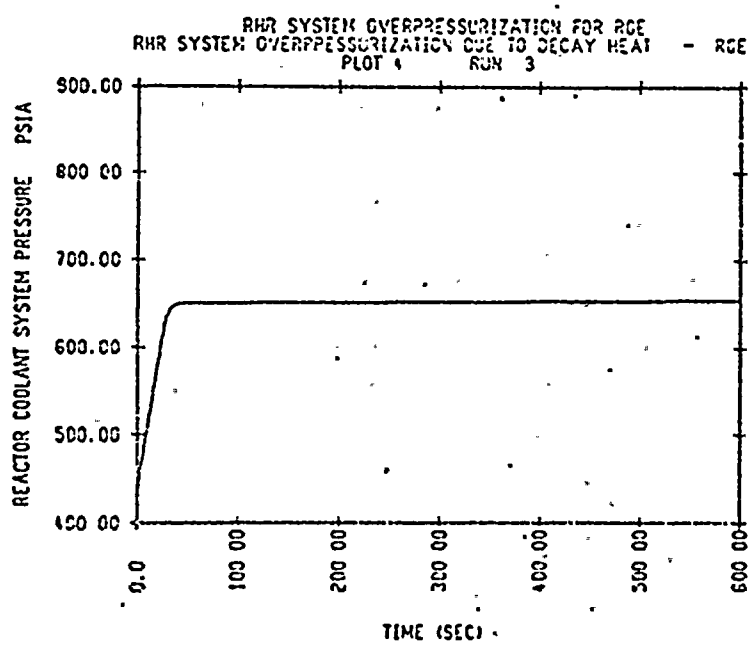
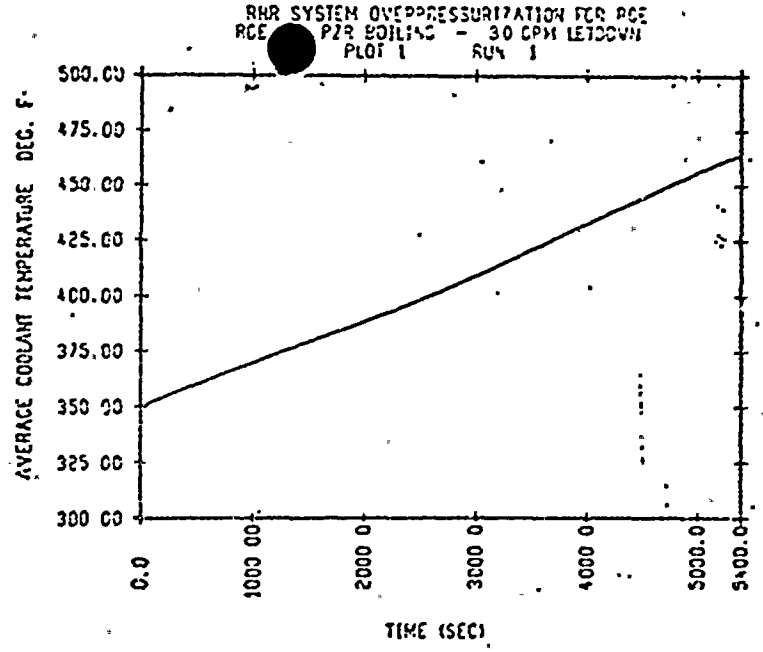
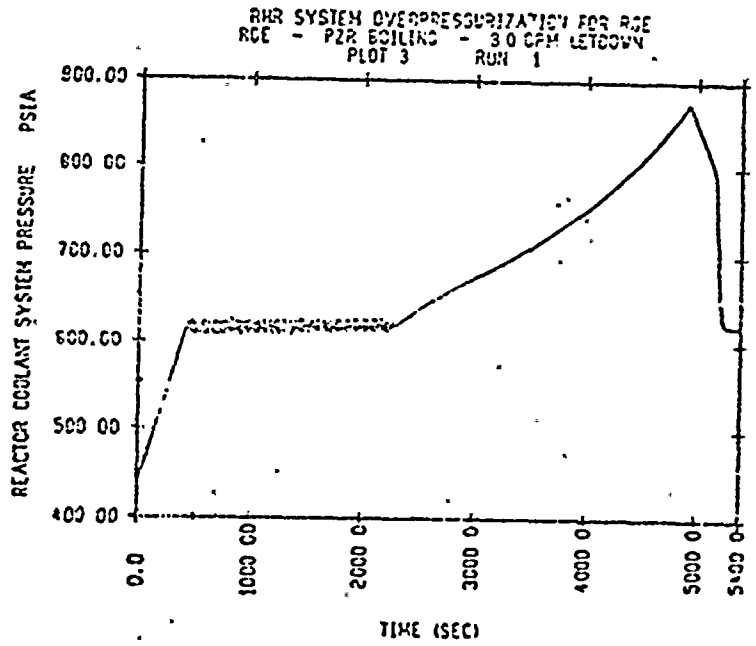
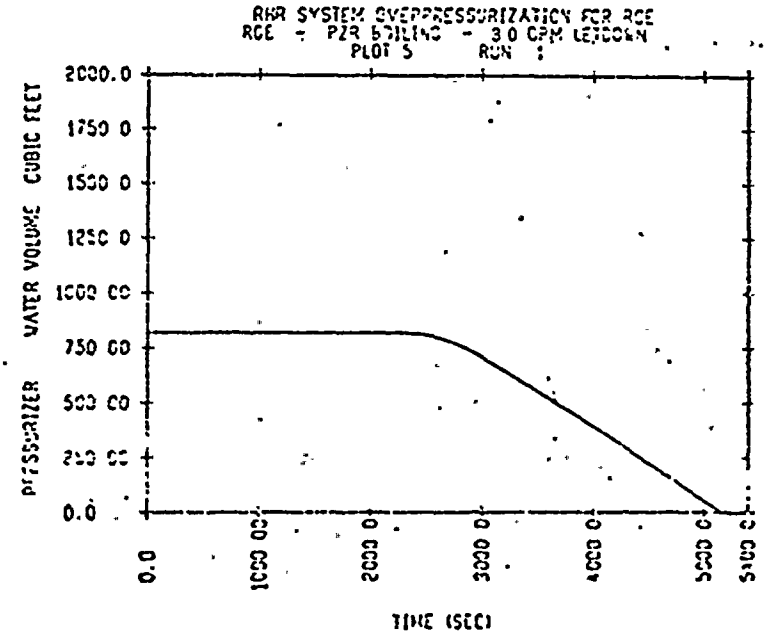

RHR SYSTEM OVERPRESSURIZATION FOR RCE
RHR SYSTEM OVERPRESSURIZATION DUE TO DECAY HEAT - RCE
PLOT 4 RUN 3
REACTOR COOLANT SYSTEM PRESSURE PSIA
900.00
800.00
700.00
600.00
500.00
400.00
0.0
100.00
200.00
300.00
400.00
500.00
600.00
TIME (SEC)

FIGURE 5







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