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February 17, 1987
Fort St. Vrain
Unit No. 1
P-87055

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Attention: Mr. H. N. Berkow, Director
Standardization and Special
Projects Directorate

Docket No. 50-267

SUBJECT: Additional Information for Analysis of Firewater Cooldown for 82% Power Operation

REFERENCE: 1) NRC Letter Heitner to Williams, dated February 3, 1987 (G-87031)

2) PSC Letter Warembourg to Berkow, dated December 30, 1986 (P-86683)

3) PSC Letter Williams to Berkow, dated January 15, 1987 (P-87002)

Dear Mr. Berkow:

In Reference 1 the NRC requested that PSC provide additional information concerning the firewater cooldown from 82% of full power in addition to that which was presented by PSC in References 2 and 3. Attachment 1 contains PSC's responses to the NRC's request for additional information.



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The Reference 2 letter submitted analyses (GA Report 909269, Issue N/C) to support FSV plant cooldown using Safe Shutdown or Appendix R Shutdown Equipment from reactor power levels above 82%. This letter forwards Attachment 2, GA Report 909269, Issue A, which is a revision of that analysis including additional information.

Reference 2 provided detailed supporting analysis only for the most limiting Appendix R shutdown cooling water flowpath (Train A). To complete the documentation, Train B of the Appendix R shutdown cooling water flowpaths has now been analyzed in detail and has been included in GA Report 909269, Issue A. Additionally, the Train A Appendix R Model has been revised to reflect a more probable operating point on the condensate pump performance curve. This point results in a lower condensate flow rate and a higher discharge pressure. The effect of this change is that the heat removal rate remains approximately the same, but less water inventory would be required to supply the condensate pump during the initial 5 hour open loop cooling configuration. The power level which can be supported by Train A remains unchanged at 83.2%.

As a result of the discussions with the NRC staff, additional data on core region peaking factors and orifice coefficients has been added to Appendix B of GA Report 909269, Issue A. In addition to the above changes, minor corrections to the text and figures of this report have been incorporated. The analysis values (Table 4-1 of GA Report 909269, Issue A) for the Environmentally Qualified (EQ) cooling water flowpath model have not been changed and continue to support cooldown from 87.5% power. The original conclusions and maximum power levels justified by the analysis remain unchanged.

If you have any questions, please contact Mr. M. H. Holmes at (303) 480-6960.

Very truly yours,

Lawrence Brey

H.L. Brey, Manager
Nuclear Licensing and Fuels Division

HLB/AHW:jw

Attachments

NRC Request 1:

Attachment 4 to P-87002, page 13, describes certain repairs and manual actions that might be required in the event that one loop is incapacitated by a high energy line break and there is a coincident single failure of the power supply for the other loop. Describe the nature of the repairs and associated manual actions. Also, provide a target date for complete implementation of this process including procedures, training and physical plant modifications.

PSC Response 1:

Safe shutdown cooling is accomplished by using either one of the two cooling loops to provide the means for decay heat removal and safe shutdown of the reactor. The essential equipment to support safe shutdown cooling is powered from the two redundant Emergency Diesel Generator (EDG) sets which are loop associated.

The scenario of concern is the occurrence of a High Energy Line Break (HELB) in a location which causes the loss of use of an Economizer-Evaporator-Superheater (EES) section for Safe Shutdown Cooling, compounded by a single failure causing loss of the primary 480 VAC Essential Bus (either Bus 1 or Bus 3) associated with the unaffected loop. This would result in loss of the power supply to two of the three helium circulator bearing water pumps or all three bearing water pumps (worst case) in the helium circulator auxiliary system of the loop unaffected by the HELB. Operation of two out of three helium circulator bearing water pumps is necessary to adequately cool and lubricate the bearings of a helium circulator in the unaffected loop following a HELB since the backup bearing water system is not environmentally qualified and may not be available. Therefore, it was necessary to identify an alternate means to power the helium circulator bearing water pumps to address the postulated circumstance.

Figure 1-1 provides a flow chart of the actions required to respond to the above described scenario. Since the scenario is identical for either loop, only the conditions associated with a single loop will be addressed in further detail (conditions 1 and 3 or conditions 2 and 4). It should be noted that all required manual actions to respond to the combination of events are confined to a mild environment (i.e., the 3-room complex).

Condition 1 is the failure of the Essential Bus 1 due to a breaker failure, insulator failure, EDG set failure or any other malfunction causing loss of power to that bus. The required actions are to diagnose and determine the cause of the Bus 1 failure, and if it cannot be re-energized in a timely manner, then the following interim measures are taken:

* Remove the "pre-fabricated" cable package from its secured location. This package consists of two cables (each with

three conductors, size 00 or larger, approximately 50 ft. long), connectors, bolts and required tools.

* Rack out the supply breakers for helium circulator bearing water pumps P-2101 and P-2106 (switchgear compartments N6 and N9 of 480 VAC Essential Bus 1) to the disconnect position.

* Pull one of the two cables from compartment N6 of Bus 1 to compartment N21 of Bus 2 (spare compartment). Pull the other cable from compartment N9 of Bus 1 to compartment N30 of Bus 2. Connect the cables to the load side of the breaker cubicle in each bus. This configuration utilizes Bus 1 as a junction box to provide the required power connection to the pumps.

* Remove the racked-out breakers from N6 and N9 (Bus 1) and install in compartments N21 and N30 (Bus 2). Trip the supply breakers in compartments N17 and N18 of Bus 3 which feeds bearing water pump P2107 and P2102 of the incapacitated loop. Close the breakers in compartments N21 and N30 of Bus 2.

  NOTE: The above actions will assure operability of all 3 bearing water pumps even though only 2 pumps are required.

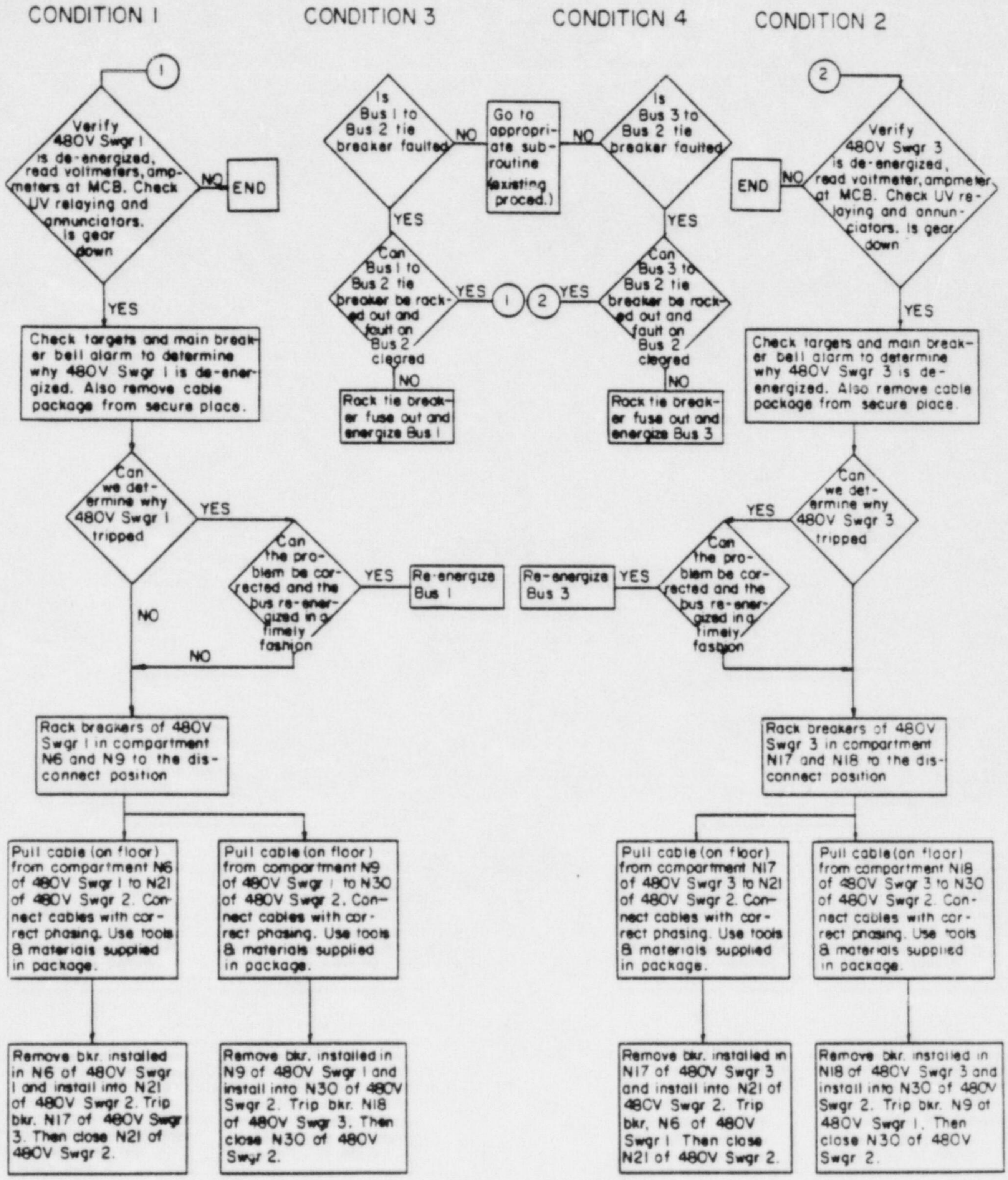
Condition 3 is the gross failure of the bus tie breaker between 480 VAC Essential Buses 1 and 2. During operation with the EDG sets, the bus tie breaker fuse is not designed to interrupt for faulted conditions. This gross failure could result in both Buses 1 and 2 being de-energized (worst case). Again, the required actions are to diagnose and determine the cause of bus tie failure. If re-energization or fault clearing can not be achieved through existing procedures in a timely manner, then the following action is required:

* Rack out the bus-tie breaker fuse. This isolates the bus-tie-breaker from Bus 1.

* Re-energize Bus 1 to supply power to the required bearing water pumps.

The procedure, procedure training and the required cable package will be in place by March 1, 1987. It should be noted that the final procedure may differ slightly from the flow chart but the required manual actions are still the same.

FIGURE 1-1




| CONDITION 1 | CONDITION 3 | PACKAGE REF'D SHALL INCL. | NOTES |
|---|---|---|---|
| (A) HELB LOOP 2<br>(B) Think we lost 480V Swgr 1 | (A) HELB LOOP 1<br>(B) Think 480V Bus 1 to 2 tie bkr. is faulted | 1) Wrenches (8 eight)<br>2) Racking tools (2 two)<br>3) 00 Cable or larger with flat connectors (8 eight) ~ 50 ft.<br>4) Bolts<br>5) Torque Wrench | 1) Cables and bus work shall be taped Red, Black, and White for AØ, BØ, CØ |
| **CONDITION 2** | **CONDITION 4** | | |
| (A) HELB LOOP 1<br>(B) Think we lost 480V Swgr 3 | (A) HELB LOOP 2<br>(B) Think 480V Bus 3 to 2 tie bkr. is faulted | | |

NRC Request 2:

It is our understanding that your analysis assumes that the liner cooling system does not operate during the 90-minute loss-of-forced-circulation cooling, assumed for a firewater cooldown. Provide an analysis that demonstrates that no damage will occur to the upper head liner during this period as a result of heat transfer from the reactor core.

PSC Response 2:

Electrical equipment of the PCRV liner cooling system is not environmentally qualified to the requirements of 10 CFR 50.49. Therefore, no credit is taken for its operation following an EQ Design Basis Event which creates a harsh environment in the Reactor Building. Attachment 3 to this letter, GA Document No. 909041 Issue N/C analyzes the effects of no PCRV liner cooling on the PCRV liner and concrete during a 90-minute Interruption of Forced Circulation (IOFC) followed by a firewater cooldown (Safe Shutdown Cooling) from 105% reactor power. Also, Attachment 4 to this letter, is GA Document No. 907935 Issue A, analyzes the effects of no liner cooling during Safe Shutdown Cooling on orifice valve temperatures and fuel temperatures. GA Document No. 907935 is the basis for GA Document No. 909041. GA Document No. 909041 concludes that during the 90-minute IOFC the highest temperature occurs at the top head liner and the concrete adjoining the liner which reach a maximum temperature of 239 degrees F at 2.05 hours into the transient. This is below the 1000 degrees F allowable liner temperature stated in FSAR Section D.1.3.1.8 and also below the 600 degrees F ASME code* concrete limit for a faulted condition when pressurized. Based on these temperatures it was concluded that the PCRV liner and concrete will continue to perform their safety functions throughout the accident, even with a loss of liner cooling for an indefinitely long period of time.

Following issuance of GA Document No. 909041, analyses were performed for Safe Shutdown Cooling from 87.5% reactor power and 10 CFR 50 Appendix R Fire Protection shutdown cooling from 83.2% reactor power levels using different secondary coolant heat removal rates than those used in GA Document No. 909041. The results of these analyses were submitted to the NRC in a letter dated December 30, 1986, Warembourg to Berkow, P-86683. GA Technologies re-evaluated the effects of no liner cooling using these shutdown models.

GA Document No. 909041 concluded that the PCRV would remain below the 600 degrees F ASME code allowable for a pressurized fault condition, provided the helium temperature exiting the circulator did not exceed 400 degrees F and the helium flow was at least 2%. The helium flow reaches a minimum of 1.5% for a period of time during EES cooldown from 87.5% power following a HELB. The helium temperature exiting the circulator, sweeping and cooling the walls of the PCRV, for this condition is approximately 100 degrees F. Using the same method of analysis as used in GA Document No. 909041, GA has made a conservative estimate and concluded that the PCRV concrete will remain well below the 600 degrees F ASME code* allowable following a HELB with Safe Shutdown Cooling (EQ cooling path) from an initial power level of 87.5%.

For a 10 CFR 50 Appendix R Train A cooldown from 83.2% power, the helium flow reaches a minimum of 1.4% of rated flow. This Appendix R cooldown with Train A follows a 90-minute IOFC with restoration of forced circulation using condensate supplied to one circulator and one EES section in the same loop. The helium temperature exiting the circulator, sweeping and cooling the walls of the PCRV, is approximately 100 degrees F. Using the same method of analysis as used in GA Document No. 909041, GA has made a conservative estimate and concluded that the PCRV concrete will remain well below the 600 degrees F ASME code* allowable during the Appendix R cooldown with Train A from an initial power level of 83.2%.

Based on the above, it is concluded that the PCRV concrete will remain below the 600 degrees F ASME code* allowable for a pressurized faulted condition following a HELB from 87.5% power, with the liner cooling system inoperative for an indefinitely long time. Also, it is concluded that the PCRV concrete will remain below the 600 degrees F ASME code* allowable following an Appendix R event from 83.2% power, with the liner cooling system inoperative for an indefinitely long time.

* ASME Boiler and Pressure Vessel Code, Section III, Division 2, Table CB-3430-1

NRC Request 3

Provide a description of the general strategy that will be used by the operators to maintain the correct helium flow to achieve the desired subcooling margin at the steam generator outlet and yet not exceed the fuel failure temperature in the reactor core.

PSC Response 3

Although the FSAR uses 2900 F as a conservative fuel temperature limit for transients, it is not a "fuel failure temperature". Temperatures well in excess of 2900 F may be withstood for short periods without rapid deterioration of the fuel particle fission product barrier (References 1, 2, and 3).

The general strategy that will be used by the operators to maintain the correct helium flow to maintain subcooled firewater at the steam generator outlet and yet not exceed the conservative 2900 F FSAR fuel temperature limit has not been changed from that previously used for shutdown cooling following various accidents analyzed in the FSAR. However, system modifications (CN 2397, CN 2412 and CN 2537) and Overall Plant Operating Procedure (OPOP) XII (in preparation), "Recovery from an Actuation of the Steam Line Rupture Detection/Isolation System", will enable the operators to overcome the recently uncovered inadequacies of Safe Shutdown Cooling reported in LER 86-026 dated August 17, 1986 (P-86587) without exceeding the 2900 F FSAR fuel temperature limit.

The system modifications in conjunction with procedure OPOP XII implement the results of analyses by GA Technologies and Proto-Power Corporation that are reported in Attachment 2 and Attachment 5. Attachment 5 is PSC's Report EE-EQ-0023, Revision A (prepared by Proto-Power). Attachment 8.1 of Attachment 5 outlines the recovery procedure that is being implemented in OPOP XII and provides the strategy to cool down the reactor core following a steam line rupture.

Important considerations developed in the general strategy for controlling primary coolant helium flow in order to prevent exceeding the 2900 F FSAR fuel temperature limit are: (1) maintenance of an adequate firewater subcooling margin and (2) the ability of the operators to control helium circulator speed. Prior to the restart of a helium circulator (90 minutes after the HELB occurs), firewater will be admitted to cooldown the EES section of a steam generator. Initially, the firewater exiting the steam generator will be steaming. After several minutes, when the firewater temperature at the steam generator outlet has decayed to about 165 F, primary coolant flow will be re-established. At this time, the firewater is 143 F below the 308 F saturated steam temperature for its controlled pressure of 64 psig. This provides a sufficient margin for increasing the temperature of the firewater exiting the steam generator by slowly increasing primary coolant flow with a helium circulator without inadvertently exceeding 308 F which would cause boiling of the firewater. Based upon operating experience, the speed of a helium circulator can be controlled (with either open or closed-loop control circuitry) within 20 rpm of the desired speed. Variations in circulator speed of this magnitude at expected circulator operating speeds in the range of 750-1100 rpm will have a negligible effect upon the helium flowrate and resultant coolant core outlet temperature. This will allow the operators to match circulator speed with the 257 F firewater temperature (specified in Attachment 8.1 of Attachment 5) at the outlet of the steam generator where there is still a 51 F subcooling margin to boiling. The 257 F steam generator exit temperature, controlled by varying the primary coolant helium flow with one helium circulator, represents the mean module temperature that provides sufficient subcooling margin (including instrument inaccuracies) to prevent boiling in the hottest modules throughout the cooldown transient. Since the procedural guidance provided by OPOP XII assures that boiling will not occur, the secondary coolant flow rates utilized in the analyses documented in Attachment 2 of this letter, and the associated heat removal rates would be expected to be achieved. Fuel temperatures remain below the conservative 2900 F FSAR limit for all cases analyzed in Attachment 2.

A concern exists as to the sensitivity of fuel temperatures to variations in the firewater temperature exiting the steam generator. That is, maintenance of a firewater exit temperature slightly below that utilized in the analyses of Attachment 2 would result in a slightly lower heat removal rate and possibly higher fuel temperatures. GA Technologies is currently evaluating the effect of controlling lower subcooled firewater temperatures on fuel temperatures. The results of this analysis will be forwarded to the NRC when the documentation is completed.

PSC considers that this general strategy will succeed in preventing fuel failure.

## REFERENCES FOR PSC RESPONSE 3

1. PSC letter dated July 24, 1979 (P-79157), Swart to Speis (NRC); Subject: Fort St. Vrain Fuel Particle Coating Failure.

2. Lunsford, J.L. {et. al.}, "Experimental and Statistical Investigation of Thermally Induced Failure in Reactor Fuel Particles," Los Alamos National Laboratory Report NUREG/CR-1787, LA-8547-MS, October, 1980.

3. PSC letter dated December 10, 1985 (P-85460), Walker to Berkow (NRC); Subject: Confirmatory Actions in Support of 35 Percent Power Restrictions During EQ Schedule Extension Period; Attachment 3: FSV Fuel Performance Under High Temperature Conditions.

NRC Request 4

Provide your target date for completion of the confirmatory review of other FSAR analyses to verify the adequacy of the steam generator heat transfer sections.

PSC Response 4

PSC has completed the confirmatory review of other FSAR Analyses to verify the adequacy of the steam generator heat transfer sections. The results of this review were submitted to the NRC by PSC letter P-87053, Confirmatory Analyses for Reactor Cooldown from 83.2% Power for Various FSAR Accidents, dated February 6, 1987. This review showed that the steam generator heat transfer sections and secondary coolant system flowpaths are adequate for decay heat removal following operation at power levels at least up to 83.2% power.

NRC Request 5

Are orifice valves reset to equal-flow positions prior to starting the circulator for cooldown?

PSC Response 5

The orifice valves are not reset prior to restarting the circulator since the analyses take no credit for any actions to adjust orifice valves after the start of the accident. The orifice valve positions and region peaking factors used in the analyses have been included on pages B-2 and B-3 of Issue A of GA Document 909269, which is included with this letter as Attachment 2.

NRC Request 6

Provide a copy of Proto-Power Calculation No. 82-09.

PSC Response 6

Proto-Power Calculation No. 82-09 is included as Attachment 6 to this letter.

NRC Request 7

Provide a basis for 125 GPM flow to booster pumps. This basis should provide a flow calculation for the water turbine train, losses through the water turbine and performance curves for the booster pumps.

PSC Response 7

The booster pump provides boosted firewater to the circulator water turbine. The firewater flow analysis, Proto-Power Calculation 82-03 Rev. B, accounts for flow to the booster pump concurrent with flow to the steam generator. In the model, firewater flow in the emergency condensate header to this take-off point is modeled as the sum of the flows to the booster pump and to the steam generator. The booster pump flow rate affects the total firewater pump flow rate and the cooling water flow rate to the steam generator. The following results of a hydraulic analysis (Proto-Power Calculation 82-18 Rev.-) of the firewater flow path demonstrate this relationship, and show that the flow rate to the steam generator is not significantly sensitive to booster pump flow. Minor changes in flow to the steam generator due to changes in flow to the booster pump will also have an insignificant effect on decay heat removal.

| Water Turbine Flow, GPM | Steam Generator Flow, GPM |
|---|---|
| 125 | 948 |
| 150 | 942 |
| 160 | 940 |

The 125 GPM flow rate to the booster pump utilized in the analysis was rounded off and is conservatively higher than the design flow rate of 121 GPM specified in the circulator technical manual, PSC O&M Manual 21-C-01-0002 (GA Technologies Report GA-A10349), for Safe Shutdown Cooling with firewater. Table 3-9 of this manual states that 121 GPM would result in a helium flow rate of 33 lb/sec, at normal PCRV inventory. Circulator performance curves and data contained in this manual are based on performance testing of circulators conducted by GA Technologies. GA Technologies Document No. 909269, Issue A, which demonstrates Safe Shutdown Cooling with firewater, specifies an initial helium flow rate of 15 lb/sec at the resumption of forced circulation, increasing to approximately 37 lb/sec 6 hours after resumption of forced circulation. The 37 lb/sec helium flow is obtained with a differential pressure at the water turbine nozzle of 175 psid. The approximate boosted firewater flow for this condition is 140 GPM to the water turbine drive. The GA Technologies' analysis conservatively utilizes a constant 940 GPM steam generator flow for secondary coolant heat removal. This corresponds to an approximate available water turbine flow of 160 GPM. The 160 GPM flow available to the water turbine drive exceeds the 140 GPM necessary to achieve the 37 lbs/sec primary coolant flow

rate. Therefore, both the booster pump flow and steam generator flow used in the analysis are conservative.

Flow calculations for the water turbine train have not been performed as part of the EQ Program. As discussed in PSC Response 8, surveillance testing is performed to ensure that the circulator speed required to support Safe Shutdown Cooling is attainable using boosted firewater to the circulator pelton wheel. This testing, coupled with initial performance testing of the circulators operating on water turbine drive, demonstrates the ability to provide the required core cooling flow rate.

NRC Request 8

Provide summary documentation of the theory, assumptions and results of the GA "alternative calculation" used as a verification check as required by 10 CFR 50, Appendix B. Summarize any supporting data from tests of the firewater or "simulated firewater" system.

PSC Response 8

Note: PSC's response has been divided into two parts with Part 1 responding to the first sentence and Part 2 responding to the second sentence of the above request.

Part 1

10 CFR 50 Appendix B under III. "Design Control" states that "The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." A combination of the different means for compliance to 10 CFR 50 Appendix B, "Design Control" has been utilized and is relevant to the current Fort St. Vrain Safe Shutdown Cooling and 10 CFR 50 Appendix R cooling analyses as follows:

Design Reviews

In late 1970 an outside firm, Jaycor, was contracted by GA to perform an independent review of both the RECA and TAP codes. The findings of that independent review were incorporated into the codes. GA subsequently prepared and submitted for NRC approval Licensing Topical Reports, References 1 and 2, for the TAP and RECA codes.

NRC subsequently determined the RECA code to be acceptable for the specific analyses performed for the Fort St. Vrain plant. The following is quoted from the NRC safety evaluation attached to Fort St. Vrain License Amendment No. 23 which states the basis for the determination of acceptability. "All reanalyses were performed using the RECA3 code: This code was not used to perform any previous analyses submitted to the NRC (i.e., for the FSAR). While the staff has not reviewed the code for applicability on a generic basis, we have determined the code to be acceptable for the specific analyses performed for the Fort St. Vrain Plant. The staff has determined the acceptability of the applicant's analysis methods by (1) evaluation of key input assumptions to which the output is sensitive, (2) comparison of the results of applicable plant transient temperature data to temperature predictions for those transients using the RECA3 code, (3) comparison of temperatures predicted by RECA3 to temperatures predicted by ORECA, and (4) comparison of analysis code predictions to hand calculations."

Alternate or Simplified Calculational Methods

Alternate calculations were utilized where feasible in the current safe shutdown and Appendix R cooldown analyses. The water side pressure drop calculations performed and reviewed by Proto-Power Corporation were partially checked by GA using the SUPERHEAT and SNIFFS codes. SUPERHEAT verified the pressure drop in the steam generator. The SNIFFS code, which is a single phase flow code, in conjunction with hand calculations where localized steaming occurs, were used to check pressure drops in piping downstream of the steam generator.

The SUPERHEAT code was used to check the steam generator performance predicted by the TAP code.

Suitable Testing Programs

The principal computer codes used by GA in the current safe shutdown and Appendix R cooling analyses are TAP, RECA, SUPERHEAT, and HOT*MODULE. The TAP code has been modified over the years to reflect the as-built plant. Numerous comparisons between TAP predictions for both steady state operation and transients have been made. The TAP predictions show good correlation to actual recorded plant data as indicated in Reference 3. Additionally, the steam generator model used in TAP has been verified by comparison with the validated SUPERHEAT code, References 4 and 5. A number of plant scrams have been studied using the RECA code. The actual plant helium flow rate and circulator inlet helium temperatures were input to the code along with initial core region power factors, helium orifice valve coefficients and prior plant operating history to establish decay heat level. The RECA predictions for the 37 region outlet helium temperatures were then compared to measured outlet temperatures. Excellent agreement was obtained after a model for the thermocouple time constant was incorporated to modify the RECA real time prediction.

Fort St. Vrain has one instrumented steam generator module. Steady state plant operating data from this instrumented module have been utilized to validate the steady state SUPERHEAT code, Reference 4.

Test results from Reference 6 for a half scale model were used to determine the fraction of flow from a specific core region to a given steam generator module. These test data were the basis for the computer code HOT*MODULE. The hot module analysis methodology is described in Appendix B of Reference 7.

10 CFR 50 Appendix B Conclusions

From the preceding discussions it is concluded that 10 CFR 50 Appendix B requirements for Design Control are being implemented. Design reviews by others have been performed for the RECA and TAP codes. The RECA, TAP and SUPERHEAT codes have been benchmarked with plant steady state operating data and transients. The HOT*MODULE code is based upon test data from a half scale test. Where possible the results of different codes are routinely compared. Examples are steam generator performance data compared between TAP and SUPERHEAT codes and SUPERHEAT and SNIFFS code results compared to Proto-Power Corporation secondary side pressure drop calculations.

## Part 2

### Simulated Firewater Cooling

Simulation of Safe Shutdown Cooling with firewater has been performed at Fort St. Vrain. The first test performed was Test T-30 in June 1976. The test did not demonstrate either the steam generator secondary flow or the circulator performance on a Pelton drive as specified in the FSAR at that time. Subsequent Test T-30A demonstrated that a 1000 gpm liquid flow rate in the EES section could be attained. Excessive valve pressure drop was identified in some Pelton supply valves. The valves were modified and a Test RT-403 demonstrated acceptable circulator performance on Pelton drive. (The results of these tests are summarized in Reference 8.) Ultimately booster pumps were added to the system to assure 175 psid water pressure is available at the Pelton nozzle.

Testing of the water turbine drive with actual firewater flow has not been performed due to the potential for contamination (dirt) of the condensate and circulator auxiliary systems. However, as required by the FSV surveillance program, the following tests are periodically performed to verify the acceptable performance of the equipment and systems required to drive each circulator with boosted firewater for Safe Shutdown Cooling. Any performance deficiencies identified as a result of testing are corrected, and then the tests are reperformed to verify that the test acceptance criteria are met.

- Per Technical Specification SR 5.2.7.a, each circulator is tested with water supplied by the booster pump to verify that the circulator speed required to attain the maximum helium flow rate required for Safe Shutdown Cooling is achieved using throttled condensate to simulate firewater conditions at a booster pump inlet pressure of 115 psig, which is conservatively less than the calculated inlet pressure of 123 psig for Safe Shutdown Cooling with firewater. This higher inlet pressure would result in a higher maximum available flow rate to the water turbine and thus higher circulator speed and primary coolant flow rate than with tested throttled condensate.

- Per Technical Specification SR 5.2.23, the booster pumps are functionally tested to verify operability, to verify that the pump shutoff head exceeds design requirements and to verify adequate motive power to one water turbine drive in conjunction with SR 5.2.7.

- Per Technical Specification SR 5.2.10.b.4.a, separate functional testing of the valves in the firewater flowpath to the water turbine drives is performed to verify that the valves will move to their Safe Shutdown Cooling positions.

- Per Technical Specification 5.2.10 the firewater pumps are periodically tested to verify that pump performance (head/capacity) meets or exceeds design requirements.

## REFERENCES FOR PSC RESPONSE 8

1. GA-A13248 (GA-LTR-21) "TAP: A Program for Analyses of HTGR Nuclear Steam Supply System Performance Transients," by A. Bardia and R.C. Potter, dated January 30, 1976.

2. GA-A14520 (GA-LTR-22) "RECA3: A Computer Code for Thermal Analysis of HTGR Emergency Cooling Transients," by J.F. Petersen dated August 1977.

3. GA-A15159 "Fort St. Vrain Surveillance Program Transient Analysis Program (TAP) Verification," by C.C. Love and R,C, Potter dated February 1979.

4. GA Document No. 909074 N/C "Validation of FSV Steam Generator Thermal Performance Codes," by D.P. Carosella dated November 19, 1986.

5. GA Document No. 909329 N/C "Verification of the FSV Steam Generator Model in the TAP Computer Code," by D.P. Carosella, to be issued.

6. USAEC Informal Report CAMD 8625 "PSC Region 1 Flow Model (Series 2) Test Report," by W.E. Walker dated July 8, 1968.

7. GA Document No. 909113B "Firewater Cooldown Using One Reheater Module (1-1/2 Hour Delay)," by R.C. Potter dated December 22, 1986.

8. Fort St. Vrain Unusual Event Report No. 5-267/76/05A dated December 7, 1976.

NRC Request 9

Justify the absence (or lack) of transient calculations for the firewater flows.

PSC Response 9

An analysis of the firewater system flow, pressure and temperature through an EES section of the steam generator during an Interruption of Forced Cooling (IOFC) has been performed. The analysis was performed in order to predict the time required to sufficiently cool an EES section after a 90 minute IOFC such that subcooled liquid flow is established through the secondary cooldown loop. Adequate subcooled liquid flow is required at the EES outlet to prevent boiling in the EES tubes when forced helium circulation is restored. The analysis is documented in Proto-Power Calculation No. 82-10, dated 11/25/86. (Submitted to the NRC in Attachment 9 to PSC letter P-86682, dated December 30, 1986.)

The results of the cooldown analysis are shown on Figure 9-1. The analysis predicts that liquid flow through the cooldown path is achieved about 14 minutes after flow is initiated. It is also noted that an independent analysis by GA Technologies was also performed to predict EES flooding times. The GA Technologies analysis (GA Document No. 909306, Issue N/C, Draft, dated 12/23/86) reported that liquid flow is obtained through the loop 8 minutes after flow is initiated. The Safe Shutdown Cooling procedures have conservatively been based on the longer 14-minute cooldown time.

The initial metal temperatures used in the analysis ranged from about 400 degrees F in the feedwater piping, 610 degrees F to 730 degrees F in the EES section and 1000 degrees F in the main steam piping. EES metal temperatures are based on calculated values following a 90 minute IOFC after indefinite operation at 105% reactor power. Feedwater piping and main steam piping temperatures are conservatively based on fluid conditions at 100% reactor power, per Fort St. Vrain Process Flow Drawing PF-2-12, Issue A. Therefore, the analysis results are valid following operation from full reactor power.

FIRE WATER CONDITIONS
DURING EES FLOODING

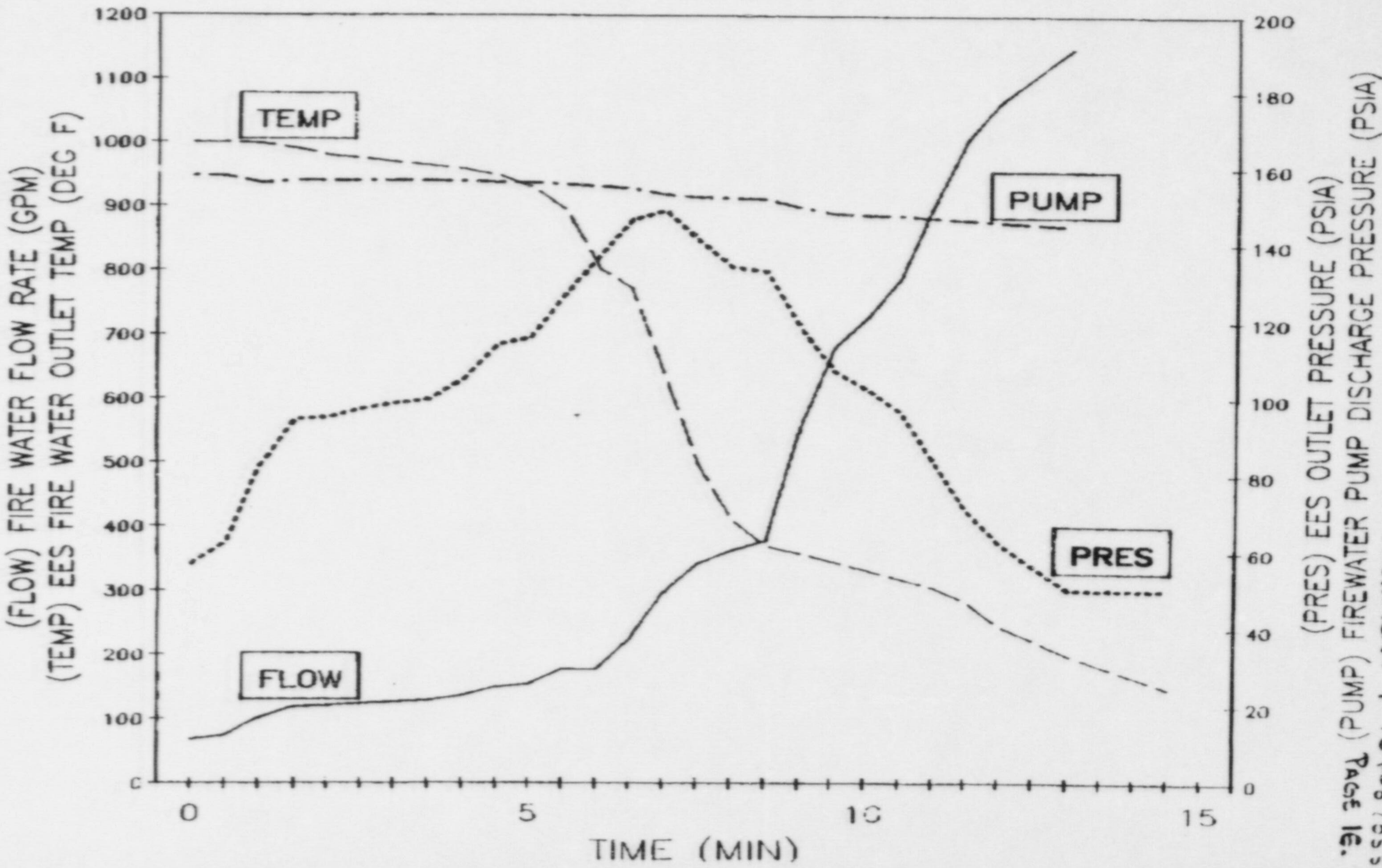



FIGURE 9-1

ATTACHMENT 1 TO P-87055
Page 16.