LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

March 18, 1981

SNRC-546

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SER REVIEW
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

The enclosed information reflects the understandings we have reached with members of your staff addressing their concerns related to the review of the Shoreham docket. This information will be formally incorporated into the FSAR at a later date.

Very truly yours,

B R McCaffrey

for J.P. Novarro
Project Manager
Shoreham Nuclear Power Station

RAH:js
attachments

cc: J. Higgins

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ATTACHMENTS TO SNRC-546

MARCH 18, 1981

1. Response to ICSB-14-223.88 ESF
2. Response to RSB-10-RCIC Set Point
3. Response to ETSB-Humidity Control
4. Response to PSB-2-Diesel Generater Reliability
5. Response to CSB-3-Purge System Duct
6. Response to MTEB-1-CRD Return Line
7. Response to ICSB-17-RPS Protection

ICSB - 14 - ESF Reset Controls

Request 223-88:

Several instances have been reported where automatic closure of the containment ventilation/purge valves would not have occurred because the safety actuation signals were either manually overriden or bypassed (blocked) during normal plant operations. In addition, a related design deficiency with regard to the resetting or engineered safety feature actuation signals has been found at several operating facilities where, upon the reset of an ESF signal, certain safety related equipment would return to its non-safety mode.

Specifically, on June 25, 1978, Northeast Nuclear Energy Company discovered that intermittent containment purge operations had been conducted at Millstone Unit No. 2 with the safety actuation signals to redundant containment purge isolation valves (48 inch butterfly valves) manually overriden and inoperable. The isolation signals which are required to automatically close the purge valves to assure containment integrity were manually overriden to allow purging of containment with a high radiation signal present. The manual override circuitry designed by the plant's architect/engineer defeated not only the high radiation signal but also all other isolation signals to these valves. To manually override a safety actuation signal, the operator cycles the valve control switch to the closed position and then to the open position. This action energized a relay which blocked the safety signal and allowed manual operation independent of any safety actuation signal. This circuitry was designed to permit the opening or certain valves after an accident to allow manual operation of required safety equipment.

On September 8, 1978, the staff was advised that, as a matter of routine, Salem Unit No. 1 had been venting the containment through the containment ventilation system valves to reduce pressure. In certain instances this venting has occurred with the containment high particulate radiation monitor isolation signal to the purge and pressure-vacuum relief valves overridden. The override of this containment isolation signal was accomplished by resetting the train A and B reset buttons. Under these circumstances, six valves in the containment vent and purge systems could be opened with the radiation isolation signal present. This override was performed after verifying that the actual containment particulate levels were acceptable for venting. The licensee, after further investigation of this practice, determined that the reset of the particulate radiation monitor alarm also overrides the containment isolation signal to the purge valves such that the purge valves would not have automatically closed on an emergency core cooling system (ECCS) safety injection signal.

A related design deficiency was discovered during a review of system operation following a recent unit trip and subsequent safety injection at North Anna No. 1. Specifically, it was found that certain equipment important to safety (for example, central room habitability system dampers) would return to its non-safety mode following the reset of an ESF signal.

In addition, any utilities do not have safety grade radiation monitors to initiate containment isolation.

SAFETY SIGNIFICANCE 223.88

The overriding of certain containment ventilation isolation signals could also bypass other safety actuation signals and thus prevent valve closure when the other isolation signals are present. Although such designs may be acceptable, and even necessary, to accomplish certain reactor functions, they are generally unacceptable where they result in the unnecessary bypassing of safety actuation signals. Where such bypassing is also inadvertent, a more serious situation is created especially where there is no bypass indication system to alert the operator.

Where the resetting of ESF actuation signals, such as safety injection, directly causes equipment important to safety to return to its non-safety mode, protective actions of the affected systems could be prematurely negated when the associated actuation signal is reset. Prompt operator action would be required to assure that the necessary equipment is returned to its emergency mode.

The use of non-safety grade monitor to initiate containment isolation could seriously degrade the reliability of the isolation system.

STAFF POSITION

It is our position that, in addition to other applicable criteria, the following should be satisfied for all operating license applications currently under review:

1) The overriding of one type of safety actuation signal (e.g., particulate radiation) should not cause the blocking of any other type of safety actuation signal (e.g., iodine radiation, reactor pressure) for those valves that have no function other than containment isolation.

2) Physical features (e.g., key lock switches) should be provided to ensure adequate administrative controls.

3) A system level annunciation of the overridden status should be provided for every safety system impacted when any override is active. (See R. G. 1.47).

4) The following diverse signals should be provided to initiate isolation of the containment purge/ventilation systems, containment high radiation, safety injection actuation, and containment high pressure (where containment high pressure is not a portion or safety injection actuation).

5) The instrumentation systems provided to initiate containment purge ventilation isolation should be designed and qualified to Class 1E criteria.

6) The overriding or resetting of the ESF actuation signal should not cause any equipment to change position.

Accordingly, you are requested to review your protection system design to determine its degree of conformance to these criteria. You should report the results of your review to us describing any departures from the criteria and the corrective actions to be implemented. Design departures for which no corrective action is planned should be justified. The following definitions are given for clarity.

a) Override: The signal is still present, and it is blocked in order to perform a function contrary to the signal.

b) Reset: The signal has come and gone, and the circuit is being cleared in order to return it to the normal condition.

Response:

Following are the responses to the Staff positions above:

1) Within the NSSS design scope, there are no overrides or one type of signal which would result in the blocking of another safety actuation signal for valves that have no other function than containment isolation. I should be noted that within the AE's design scope, one exception exists to the above design approach whereby SRV air supply inboard isolation and SRV normal air supply isolation valves are closed upon a safety injection signal, placing the SRV's on their emergency bottled air system. After approximately 10 minutes, the operator has the option, if the normal air system is present, to override isolation and revert back to the normal air supply. This will save the emergency air supply for further needs.

2) Automatic action of engineered safety features(ESF) can be overridden in two ways:

(a) Lockout functions - The lockout function is accomplished with pull-to-lock control switches. The lockout function is used for maintenance of a system or component, i.e., pump breaker. Once in "Lock-Out" the auto and remote manual controls are overridden.

(b) Manual Override of an automatic action of an ESF System - The following systems have overrides: Nuclear boiler, residual heat removal, core spray, reactor core isolation cooling radwaste, reactor building standby ventilation, control room A-C chilled water, compressed air, and control room air conditioning systems. The manual override allows the operator to secure equipment that is no longer needed or for reasons indicated on Table 223.88.

The capability to manually override safety system initiation signals after completion of accident or transient mitigation, is necessary to allow other modes of system operation. These overrides place system control in the hands of the control room operators, utilizing normal equipment controls. Keylocking of these controls would greatly reduce operator capability to respond to abnormal events.

3) All manual overrides which render a safety system inoperable during maintenance and testing are annunciated at the system level as required by R.G. 1.47.

Operator override of safety system initiation signals following accident mitigation is indicated by a light only with no annunciation since the safety function is not out-of-service.

4) Shoreham's containment purge/ventilation system is isolated upon: a) reactor vessel low water level; b) drywell high pressure; c) refueling platform level high radiation: d) reactor building low differential pressure. At present, there is no containment isolation signal for high radiation.

5) The instrumentation system provided to initiate containment purge/ventilation isolation is designed and qualified to class 1E criteria, IEEE-323 (1971) and IEEE-279 (1971).

6) Within the BOP design, for those components with override circuits that utilize their own control switches to initiate an override, the component will return to its non-safety position or function upon overriding.

Resetting - Review of ESF circuits indicated that in two instances (both in the RBCLCW/System) a reset or "LOCA" signal will return AOV's to their "pre-accident" positions for example, during normal plant operation valves 1P42*TCV0001X and W, modulate to control flow through the RBCLCW heat exchangers on system demand. This provision is made in order to protect components from too low or too high cooling water temperatures. These components being noncritical loads are isolated during an accident: also, valve TCV001X is closed and TCV001W is opened to direct full cooling water flow through the heat exchanger which is the system fail/safe position for the valves. Now, if the "LOCA" signal is reset, the valves revert to the modulating mode. However, this will not degrade RBCLCW capability during the accident, because it still meets system demand and design requirements.

Also, 1P42*AOV282, 293, 294 are testable check valves with spring assist in close direction during test. "LOCA" signal deenergizes the solenoid valve to apply spring tension on valve in close direction. Resetting of "LOCA" signal will remove spring tension, allowing valve to operate as a simple check valve. The spring assist feature is provided to minimize leakage out of the closed system if level in the surge tank drops below a predetermined level, then a "low level surge tank" signal deenergizes the solenoid valve to apply spring tension on the valve in the closed direction. This signal cannot be overridden: therefore, the RBCLCA capability will not be degraded.

Within the NSSS design, manual override of ESF actuation signals following mitigation of the event will result in equipment reverting to another position which is safe for the new mode of system operation that is entered.

Resetting of ESF actuation signals will cause the following containment isolation valves to open rather than remain closed:

a) Reactor water sample valves.

b) In addition, the RHR Heat Excanger sample line valves which are containment isolation valves during the RHR shutdown cooling mode will also reopen.

223-88d

TABLE 223.88

LIST OF MANUAL OVERRIDES (See Note 1 and 2)

Equipment (Equip. No.)	Type Override	Reason Override Provided
Supp. Pool Pump Back System Discharge Valve (1G11*MOV639)	Separate manual override switch	Provide pump discharge path to suppression pool
RBSVS & CRAC Water Chillers (1M50*WC-003A,B) (1M50*WC-004A,B)	Manual override using normal equipment control switch	Used to override auto start when equipment is no longer needed
Compressed air to SRV (Emergency Supply) inboard isolation valve (1P50*MOV105 A,B) Compressed air to SRV Normal Supply (1P50*MOV113 A,B)	Separate manual override switch	Revert from emergency to normal air supply if normal supply available
RBSVS Booster Fan (1T46*FN-079A,B) RBSVS Unit Coolers (1T46*UC-002A,B) (1T46*UC-003A,B) (1T46*UC-004A,B) (1T46*UC-005A,B) RBSVS Filter Inlet Air Dampers (1T46*MOD048A,B)	Manual override using normal equip. control switch	Used to override auto start when equipment is no longer needed
CRAC Outboard Isolation Valves (1X61*MOV31A,B) (1X61*MOV32A,B)	Manual override using normal equip. control switch	Used to isolate when radioactive contamination detected in air supply line
LPCI outboard injection valves (1E11*MOV-036A,B)	After 5 minutes time delay can manually override normal equip. control switch	No further need for LPCI injection
LPCI Pumps (1E11*P-014A,B,C,D)	Manual override using normal equip. control switch	No further need for LPCI pump

TABLE 223-88 (CONT'D)

Equipment (Equip. No.)	Type Override	Reason Override Provided
LPCI Pump Suction Valves (1E11*MOV31A,B,C,D)	Key lock switch Note 3	Transfer suction for shutdown cooling mode
Drywell Suppression Pool Spray Valves (1E11*MOV038A,B) (1E11*MOV039A,B)	Separate manual override switches	To allow Drywell or Suppression Pool Spray with LPCI injection signal present
Core Spray Pump (1E21*P-013A,B)	Manual override using normal equip. control switch	No further need for core spray pump
Core Spray Injection Valves (1E21*MOV33A,B)	Manual override using normal control switch	No further need for core spray
Core Spray Suction Valves (1E21*MOV031A,B)	key lock switch Note 3	Transfer suction during testing
RCIC Turbine (*TU005)	Manual override via Turbine Trip push-button. Requires resetting of Turbine Trip and Throttle Valve for restart.	

NOTES:

1. All 4160 Volt and 480 V equipment have Pull-to-Lock switches which are used to "Lockout" the associated equipment for maintenance purposes.

2. In case of loss of control room, selected ESF equipment has the capability of being operated from the Remote Shutdown Panel. Once in local control at the RSP, auto and manual signals through the control room or relay room are defeated.

3. Suction valves have no signals to be overridden. These are best described as administrative controls.

RSB-10 - RCIC TEMPERATURE SET POINT

"Show how the design of the RCIC protection system prevents unintentional shutdown of the system, when the system is required, because of spurious ambient temperature signals from areas in and around the system (especially in the RCIC pump room)."

Response:

The unintentional shutdown of the RCIC System has been precluded by the selection of temperature sensor locations based upon both the postulated break points in the steam piping system and the building ventilation intake points. As a result of this selection, the temperature elements measuring ambient air temperature are not located in and around the actual equipment. Also, it should be noted that the RCIC equipment is not located in a room or cubicle. Shoreham's reactor building is configured in a large open area on the elevation this equipment is located on. The RCIC system steam leak detection temperature sensors are essentially looking at an open area resembling a plenum. In addition, the temperature setpoint for alarm and system isolation has been specified as 30F above the ambient temperature. This allows a margin for localized heat up without sacrifice of response time for lead detection. Actual temperature settings will be determined during the startup test program.

ETSB Humidity Control

INSERT A

The control room air conditioning system is designed to maintain relative humidity of less than 60 percent. This is an inherent design feature of the system due to high internal sensible heat load and relatively small latent heat load (maximum of 10 people). Outside air of 500 cfm during normal operation (1,000 cfm under accident condition) does not contribute significantly to the control room latent heat load. Moreover, a control system is provided for dehumidification to ensure the design temperature and humidity is not exceeded.

INSERT B

Under design operating conditions this mixture of airstreams results in a relative humidity of the air entering the filter trains, less than 70 percent, thus precluding the need of a preheater in the filter trains.

ETSB - HUMIDITY CONTROL

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 Control Room Air Conditioning (CRAC) System

9.4.1.1 Design Bases

The control room air conditioning system provides ventilation, cooling, and control of maximum relative humidity. The air conditioning equipment is designed to:

1. Provide temperature and humidity control 75 F and 60 percent maximum, respectively, and to control the air movement for personnel comfort and equipment performance.

2. Maintain a positive pressure above atmospheric pressure to prevent air leakage into the main control room during isolation.

3. Have the capability of shunting the supply air to the main control room area through a special filter train containing charcoal filters during accident or main control room isolation modes.

4. Have the capability of operating during normal, shutdown, and design basis accident (DBA) conditions without loss of function.

5. Provide radiation monitoring of outside air supply.

The main control room air conditioning system and chiller room ventilation system are safety related and designed to Seismic Category I requirements. The toilet and kitchen exhaust fan is not safety related.

9.4.1.2 System Description

The main control room is served by two 100 percent capacity redundant air handling units (one operating and one spare). Air for the control room kitchen and toilet areas is also provided by this system (Fig. 9.4.1-1). Each unit consists of a filter assembly, cooling coil, fan, and dampers. The units discharge air into a common duct, then to the distribution ductwork and supply registers in the area. Air intakes and exhausts are tornado missile protected. The supply air is cooled by chilled water cooling coils. Two redundant chilled water supply systems are provided as described in Section 9.2.9. The kitchen and toilet area is exhausted by a separate fan and exhaust duct system.

INSERT A →

To filter the control room outside air supply, two 100 percent standby air filter trains are provided with one serving as a spare. Each filter train consists of the following principal components:

1. Filter cabinet assembly consisting of steel frame, filter mounting structures, access doors, covering, and supports.

2. Dry media type prefilter.

3. High efficiency particulate air (HEPA) filter bank located downstream of the prefilter.

4. Charcoal filter bank for radio-iodine absorption.

5. Final HEPA filter bank located downstream of the charcoal filter bank for removal of charcoal fines from the discharge air stream.

6. Filter instrumentation consisting of flow and pressure differential switches, transmitters, and alarms to facilitate monitoring and testing filter operation.

The HEPA filters have a minimum efficiency of 99.97 percent by dioctylphthalate particulate (DOP) testing and are capable of operating at 250 F and 100 percent relative humidity.

The charcoal filter bed is capable of removing 95 percent of airborne radioactive iodine (I) and 95 percent of methyl iodine at the design flow of 4,000 cfm. Booster fans are included in the system to overcome the increased resistance of the filter trains.

During normal operation, supply air is filtered by prefilters and 50 percent minimum average dust spot efficiency filters (National Bureau of Standards atmospheric). In addition, during an accident condition, provision is made to route a mixture of 1000 cfm outside and 3000 cfm recirculated air through the charcoal filter trains. INSERT IB

Two separate outside air intakes are furnished to provide alternate sources of outside air. The intakes are located so that one of the intakes continually ensures clean air for the control room pressurization. One intake is located on the west wall of the control room building, and one intake is located on the east wall of the turbine building. The air intakes are located approximately 450 ft apart, and approximately 160 degrees apart in relation to the plant exhaust and offgas vents. The air intake located on the east wall of the turbine building is shown on Fig. 9.4.1-2. This duct is 14 in. standard weight carbon steel pipe, welded for leak tightness, up to isolation valve MOV 031A and MOV 032A 1B (Fig. 9.4.1-1) located in the HVAC equipment room. This duct is routed through the turbine building at el 63-0. The pipe is seismically analyzed.

During accident conditions, one of the redundant control room standby air filter train booster fans will be started, and outside air will be automatically routed through the charcoal

REQUEST 223.79 (9.4.10.3):

Expand the discussion on safety evaluation in Section 9.4.10.3 to include possibility of the accidental release of any of the stored gases mentioned in Table 9.5.9-1 and the possible affects on diesel generator operation such as reduced power capability and difficulty in starting.

RESPONSE

The response to this request is incorporated in FSAR Section 9.4.10.3.

thermostat, located at the exhaust air outlet, modulates the dampers to maintain desired room temperature. Each supply fan runs when its associated diesel is running and stops when its associated diesel is stopped. Each fan is powered from the normal power supply or, on loss of offsite power, from the emergency bus associated with its diesel.

Normally, when a diesel is not running, ventilation is provided by a separate 1350 cfm exhaust fan and associated exhaust air fire damper for each room, sized for 1 cfm/ft^2 of floor area. Supply air is provided by infiltration.

Heating for each room is provided by electric unit heaters controlled by a room thermostat. Unit heaters will not be running when diesels are running, as waste heat from the diesels will maintain minimum room temperature.

9.4.10.3 Safety Evaluation

The diesel generator supply fans, ductwork, and dampers are designed to Seismic Category I requirements. The ventilation systems are tornado missile protected. Each ventilation system is provided with standby power from the emergency diesel generator which it serves. Redundancy for single failure is achieved through three separate ventilation systems. Thus, if a ventilation system should fail and its associated diesel, at the discretion of the control room operator, has to be shut down due to high ambient temperature, the two remaining diesels have the capacity to achieve a safe plant shutdown.

The electric unit heaters and the normal ventilation exhaust fans are not safety related.

The fire protection system for the diesel generator room is provided by a CO_2 total flooding system. Should an accidental discharge of CO_2 occur, the ventilation system will continue exhausting the air and fumes at the opposite end of the control building from the air intake. If CO_2 is initiated either manually or automatically, the fans will stop and will not start again unless fire protection initiation logic is reset in the main control room. When reset, fans will start automatically. Additionally, combustion air for the diesel engine is ducted from the outside air intake directly to the engine and therefore, diesel operation is not affected by an accumulation of fumes within the room.

INSERT A

Rupture of the CO_2 storage tank will not affect emergency diesel operation due to the limited supply of CO_2 contained in the tank.

The combustion air intake and diesel exhaust are on opposite sides of the control building, approximately 80 ft apart. Additionally, the exhaust is discharged above the building roof in a westerly direction as shown on Fig. 3.8.4-7. Thus, it is highly improbable that

INSERT A

The onsite storage of gases under pressure is tabulated in FSAR Table 9.5.9-1. Of the gases listed, only carbon dioxide and propane are stored in the vicinity of the diesel generator rooms. All other gases are remotely located.

A small quantity of propane (2-100 pound bottles each containing 855 scf) is located outdoors on the east side of the control room building (auxiliary boiler room) at el 20-0. The nearest diesel generator room is located approximately 40 feet away from the propane storage. The cylinders are equipped with relief valves and the quantity of gas which would be discharged through a relief valve is small. The diesel generator room air intake hoods for both combustion and ventilation are located on the west or opposite side of the control room building and therefore would not be affected by an accidental release from the propane storage bottles.

A carbon dioxide (CO_2) storage tank is located outdoors at el 15-6 on the west side of the control room building. Outdoor air intake hoods for diesel generators 101, 102, and 103 are located at approximately 82, 32, and 57 feet, respectively, south of the CO_2 tank. The intake hoods are elevated 15.5 feet above the tank elevation. The grade at the location of the tank is away from the diesel generator rooms.

The CO_2 storage tank (12.5 tons capacity or 225,000 scf gas equivalent) is designed, manufactured, tested, and inspected in accordance with Section VIII Division 1 of the ASME Boiler and Pressure Vessel Code. The tank is equipped with a system of multiple safety devices and is monitored and alarmed in the control room for high and low pressure and for low liquid level. Since the diesel generator air intake hoods are separated from the CO_2 storage tank by a significant distance, accidental release of CO_2 will have no adverse affect on the starting or satisfactory operation of the diesel generators.

CSB-3 PURGE SYSTEM DUCT

As requested by the NRC, the operation of the containment isolation valves on the Containment Purge and vent Lines have been evaluated as they relate to the integrity of the purge system duct work outside containment. The 18-inch purge valves and the six-inch vent valves, as indicated on the FSAR Table 6.2.4-1, are in a normally closed position. Thus, a DBA would not pressurize the duct system. There are, however, as designed, times when the six-inch vent isolation valves are opened to perform the vent system function. These quick acting air operated valves (IT46*AOV078A & B) have been designed to satisfy the requirement of BTP CSB 6-4, with closure times not exceeding five seconds. It is estimated that these vent valves would be in an open position for operation no more than 60 hours per year.

The ductwork beyond the isolation valves of the primary containment venting system has been reviewed concerning the effects of escaping air and steam from the drywell caused by a design basis accident (DBA) before the isolation valves are fully shut. The ductwork in question is not required to be serviceable after an accident and is not designed to withstand the effects of a DBA pressure increase. Loss of integrity of this ductwork system will not affect any safety related equipment since there is none located in the immediate vicinity of the ductwork. This scenario has no effect on plant safety and the components affected are not required to operate afterwards. No further action is required.

CSB-3

TABLE 6.2.4-1 (CONT'D)

NOTES

11. Special air testable check valves with a positive closing feature are designed for remote testing during normal operation to assure mechanical operability of the valve disc. The remote testing feature will cause only a partial movement of the disc, into the flow stream, with only a minor effect on flow. Upon receipt of an isolation signal, the actuator spring force will either cause a slight reduction in flow when the feedwater system is available or cause the valve to close, providing a positive closure differential pressure on the seated disc, when the feedwater flow is not available.

12. This valve will open when both a low reactor pressure vessel pressure and an accident signal are present.

13. The motor operator of this valve is keylocked open during normal operating conditions.

14. Traversing In-Core Probe (TIP) System

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of four valves may be opened at any one time to conduct the calitration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, as indicated by a containment isolation signal, the cable is automatically retracted and the ball valve closes automatically after completion of cable withdrawal. To ensure isolation capability, if a TIP cable fails to withdraw or a ball valve fails to close, an explosive, shear valve is installed in each line. Upon receipt of a remote manual signal, this explosive valve will shear the TIP cable and seal the guide tube.

15. All unused penetrations (designated "Spare") are capped and seal welded.

16. Valve will close on system high flow.

ept 17. Isolation signals A or F will initiate the reactor building standby ventilation system which in turn isolates the purge air isolation valves.

18. This valve will open when both a low differential pressure across the valve and an accident signal are present.

17. ISOLATION SIGNALS B OR F WILL INITIATE THE REACTOR BUILDING STANDBY VENTILATION SYSTEM [RBSVS] AND SIMULTANEOUSLY, THROUGH A LOCKOUT RELAY ISOLATE THE PURGE AIR ISOLATION VALVES.

MTEB - 1 - CRD Return Line Modification

The information requested regarding the removal of the CRD line has been submitted for NRC Staff review via the following General Electric (GE) documents:

a. Letter of March 14, 1979, G.G. Sherwood (GE) to V. Stello and R. Mattson (NRC) regarding calculation of CRD system return flow capacity;

b. Letter of April 9, 1979, G.G. Sherwood (GE) to V. Stello and R. Mattson (NRC) forwarding results of CRD system solenoid valve endurance testing;

c. Letter of May 1, 1979, G.G. Sherwood (GE) to V. Stello and R. Mattson (NRC) forwarding results of CRD system solenoid valve performance testing; and

d. Letter of November 2, 1979, G.G. Sherwood (GE) to R.P. Snaider (NRC) forwarding additional information requested regarding CRD hydraulic system performance, especially with regard to corrosion products emanating from carbon steel piping.

e. Letter of November 27, 1979, G.G. Sherwood (GE) to R.P. Snaider (NRC) forwarding a) results of analyses of boil-off rates and CRD pump makeup capability for plants not previously addressed including the 218" BWR 4 RPV applicable to Shoreham, and b) a draft procedure for optimizing CRD pump flow to the reactor.

The above submittals have been reviewed and accepted by the NRC Staff as indicated in the following documents:

1. Letter of January 28, 1980, D.G. Eisenhut to R. Gridley regarding G.E. submittals a through d above.

2. Letter of February 11, 1980, D.G. Eisenhut to R. Gridley regarding G.E. submittal e above.

These NRC letters required certain additional modifications for plants which chose to remove the CRDRL without rerouting. The modifications applicable to Shoreham are addressed below. In addition, during preoperational testing, tests will be conducted to verify proper operation of the CRD system, to determine return flow to the reactor vessel, and to confirm the flow rate to the CRD system with concurrent two-pump operation.

NRC MODIFICATION

4a. Installation of equalizing valves between the cooling water header and the exhaust water header.

SHOREHAM RESPONSE

In order to assure satisfactory system operation with the single failure of an equalizing valve, the proposed design modification will include the addition of two equalizing valves installed between the cooling water header and the exhaust water header in a parallel configuration. The failure of either valve will not impair CRD operation for any foreseen operating or accident condition.

NRC MODIFICATION

4b. Flush ports installed at high and low points of exhaust water header piping run if carbon steel piping is retained; and

SHOREHAM RESPONSE

For Shoreham all lines are made of stainless steel.

NRC MODIFICATION

4c. Replacement of carbon steel pipe in the flow stabliizer loop with stainless steel and rerouting directly to the cooling water header.

SHOREHAM RESPONSE

For Shoreham all lines are made of stainless steel.

NRC MODIFICATION

5. Each licensee must establish readily-available operating procedures for achieving maximum CRD flow to an otherwise isolated reactor vessel.

SHOREHAM RESPONSE

Readily available operating procedures for achieving maximum CRD flow will be prepared.

NRC MODIFICATION

Provide one or two meters capable of reliable direct measurement of one and two pump flow.

SHOREHAM RESPONSE

The primary function of this system is not to provide make-up water to the RPV. In addition, this system is not classified safety-related. Accordingly, we consider that these additional meters are not required for safety.

GENERAL ELECTRIC

NUCLEAR ENERGY
PROJECTS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

March 14, 1979

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Victor Stello, Jr., Director
Division of Operating Reactors

Mr. R. J. Mattson, Director
Systems Safety

Gentlemen:

SUBJECT: CONTROL ROD DRIVE (CRD) RETURN LINE REMOVAL

References:
1) January 27, 1978 letter, G. G. Sherwood (GE) to E. G. Case (NRC), same subject
2) July 14, 1978 letter, G. G. Sherwood (GE) to V. Stello (NRC) and R. J. Mattson (NRC), same subject

The referenced letters advised the NRC of General Electric's intention to delete the CRD return line. This change results from GE's finding that all functional requirements of the CRD system can be met without this line. GE requests NRC concurrence with this change.

During subsequent discussions with NRC staff, it was agreed that the proposal should be evaluated for its effect upon the capability of the CRD pump(s) to supply high-pressure water to an isolated reactor vessel. GE performed analyses to determine water-injection capabilities of present and proposed designs, assuming in both cases that all practical valve adjustments to optimize flow (those which an operator can make from outside the primary containment) have been accomplished.

To assess the effect of this change, GE calculated the injection rate that would result for various product lines and plant sizes, and these are shown in attachments 1A, 1B, 1C, 1D, 1E and 1F to this letter. Attachment 1A is a tabulation of boil-off rates and make-up capabilities and the others are curves of water levels and time following scram for different make-up flow rates. In each sample presented, the lowest curve (designated "A") represents the minimum flow needed to keep the core covered, while the remaining curves indicate trends with higher

injection rates (typically 10 to 20 GPM increments). The values presented in attachment 1A have resulted from calculations based upon the conditions and assumptions shown in attachments 2 and 3. The values in attachments 1A through 1F and attachments 2 and 3 have been reviewed and verified since the initial calculations were presented during the May 1978 meeting with NRC staff. The verification process covered the following:

1. Calculation model of CRD injection and vessel boil-off.

2. Model input data including hydraulic coefficients, pump and valve characteristics, and decay heat.

3. Assumptions made.

4. Calculation of results.

Certain sizes and product lines were selected as representative and included in the attachments as examples. For those plants which request removal of the CRD return line, additional calculations may be necessary.

Data presented for BWR 4 and BWR 6 plants are for bounding plant sizes in either product line. Within a given product line, sensitivity to return line deletion is primarily a function of the number of control rods in a plant. Hence, the sensitivity to deletion of the return line for other plants within these product lines is expected to lie within the range given. There appears to be no reason why BWR 5 sensitivity to return line deletion is not comparable to those for BWR 4 and BWR 6.

Also, enclosed as attachments 4, 5 and 6, are copies of transparencies which have been reviewed per discussions with NRC staff (and which replace earlier versions presented at the August 1978 meeting with NRC staff). These revisions reflect the role played by CRD system flow during the repressurization phase of the Browns Ferry incident following temporary loss of relief valve function. While the material in these attachments has not been specifically verified, it appears to be acceptably accurate in view of the verification of the previous analysis.

Summarizing this information, GE concludes that the capability of the CRD system to provide water to the reactor vessel is not significantly affected by removal of the CRD return line. The line has been deleted from the BWR/6 design, and GE proposes that it should be removed from other plants as the means of reducing stress corrosion problems.

GENERAL ELECTRIC

Meanwhile, GE is reviewing the situation with the intent of advising BWR owners of operating procedures which they might follow to mitigate effect of another Browns Ferry type incident. This advice, which will be provided in a Service Information Letter (SIL) to be issued later this year, is expected to address:

1. Advantage of early vessel blowdown.
2. Water inventory to be maintained in the vessel.
3. Alignment of control rod drive system valves.
4. One and two CRD pump operation to provide maximum water flow.

Very truly yours,

Glenn G. Sherwood, Manager
Safety and Licensing Operation

GGS:mm/64-66

ATTACHMENT 1A

RELATIVE INJECTION FLOW RATES (GPM)

PLANT	ONE PUMP		TWO PUMPS**	
	Present Design	Proposed Design	Present Design	Proposed Design
218 BWR/6	180	160	240	200
251 BWR/6	180*	170	245	215
251 BWR/5	170	135	295	175
183 BWR/4	180	110	315	130
251 BWR/4	180*	165	345	205

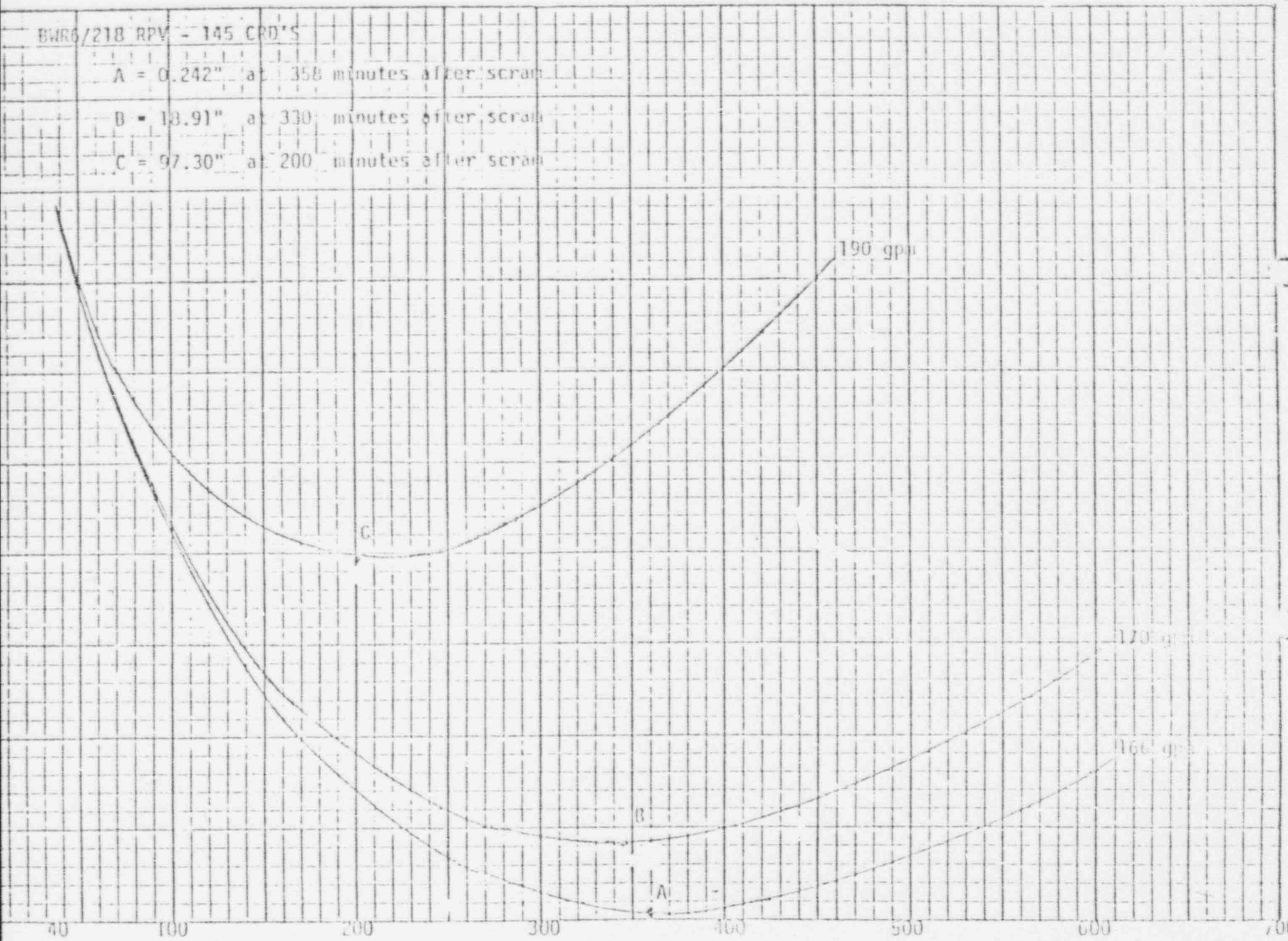
*Limited by pump runout capacity
**Not a design requirement. Assumes two pump operation is viable.

PLANT	Minimum Makeup Flow to Keep Core Covered (GPM) CRD Pump is Sole Source 40 Minutes After Scra
218 BWR/6	165
251 BWR/6	215
251 BWR/5	180
183 BWR/4	95
251 BWR/4	180

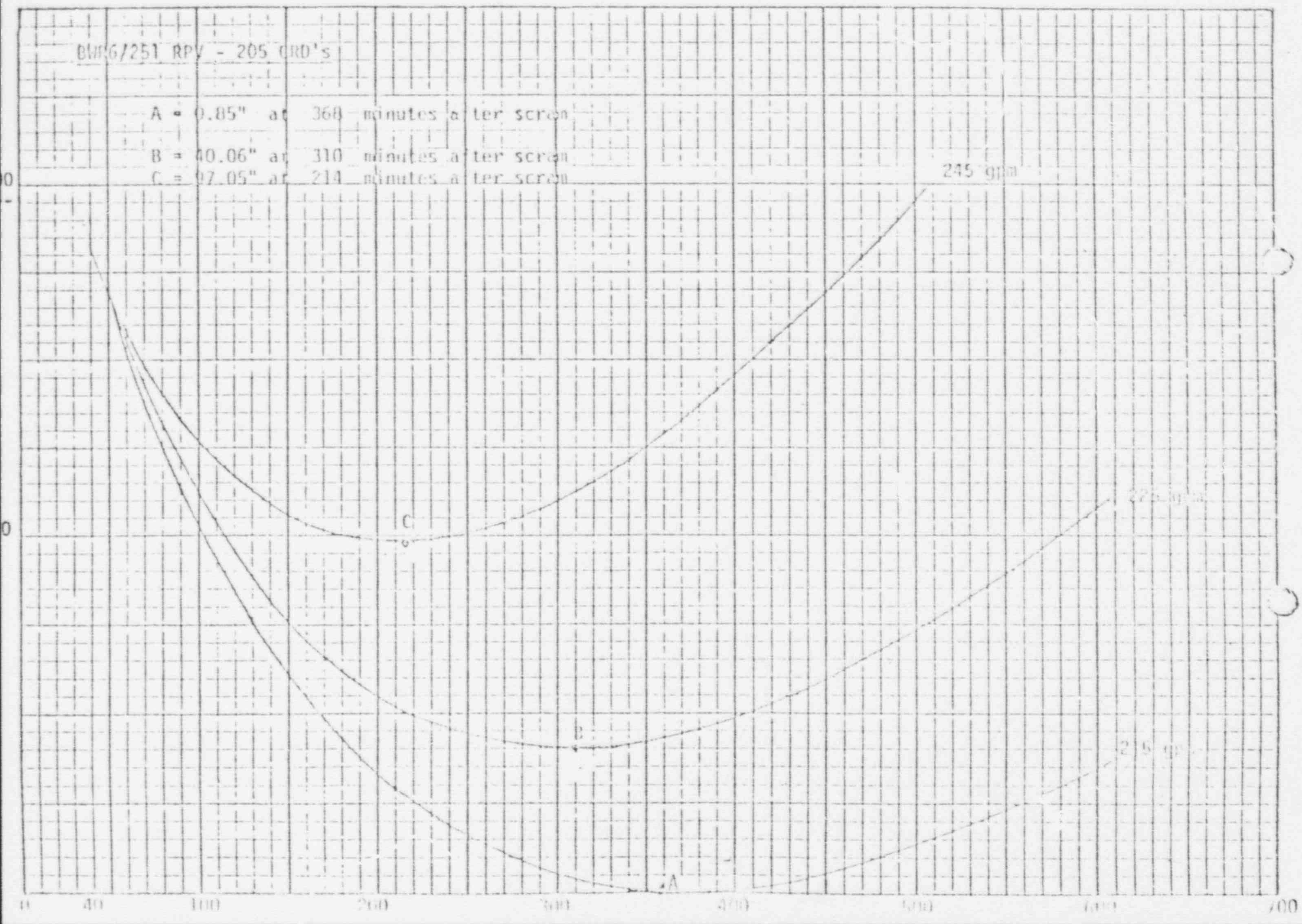
(All data have been rounded off to the nearest 5 gpm)

ALS:mm/117
3/14/79

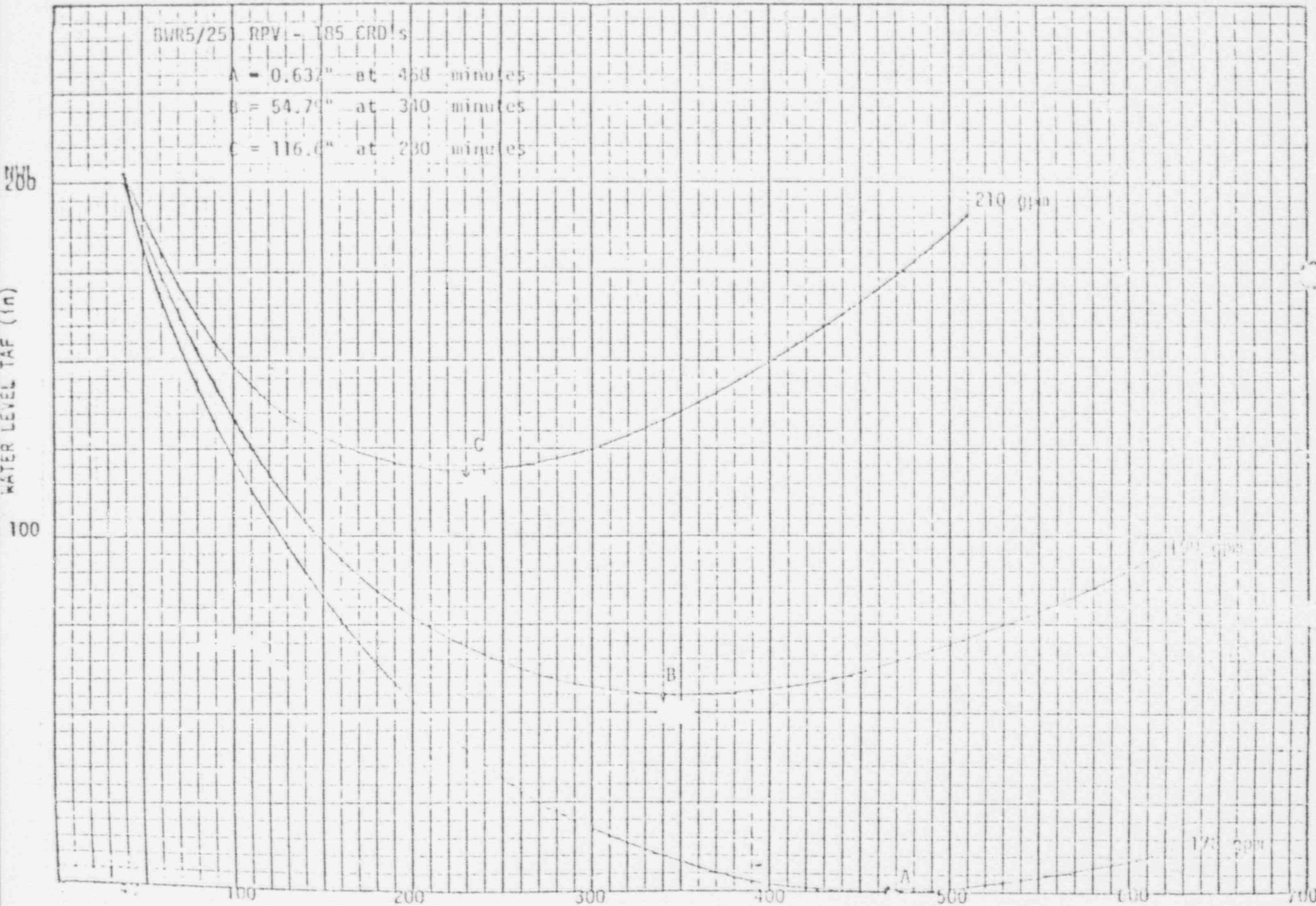
ATTACHMENT 1 B



ATTACHMENT 1 C

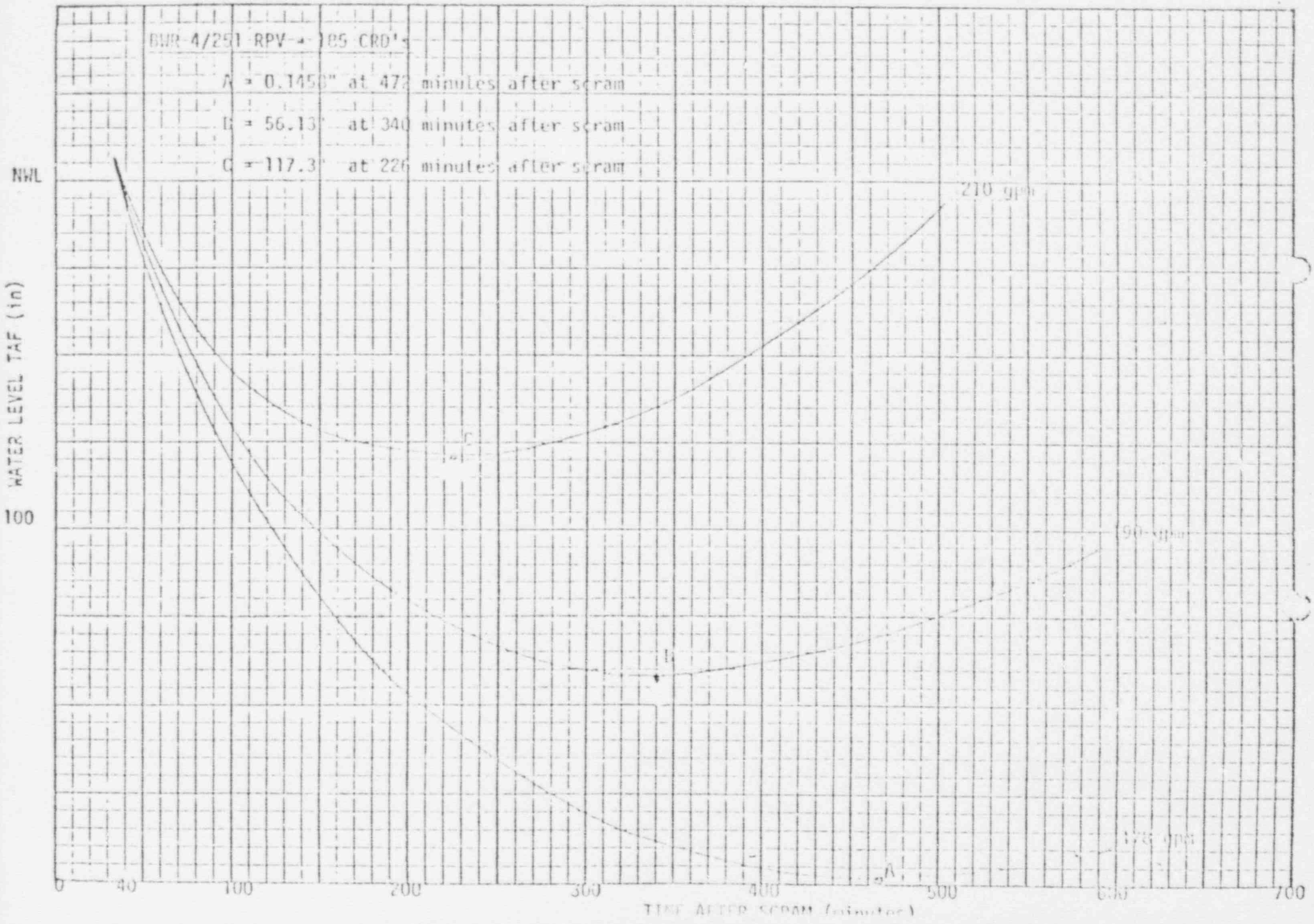


ATTACHMENT . J



ATTACHMENT 1E

ATTACHMENT 1 F



ANALYSIS OF MAKE-UP CAPABILITY
SYSTEM CONDITIONS MODELED

GENERAL (ALL PRODUCT LINES)

(1) REACTOR PRESSURE AT RELIEF VALVE SAFETY SETPOINT + 15 PSI TOLERANCE + 20 PSI STATIC HEAD

(2) SINGLE PUMP AND DUAL PUMP OPERATION WITHOUT REGARD TO POSSIBLE SUCTION PRESSURE LIMITS

(3) PUMP DISCHARGE PRESSURE ASSUMED EQUAL TO PUMP TDH AT RATED FLOWS

(4) DRIVE WATER FILTER CLEAN

THIS ASSUMPTION IS BASED ON THE EXISTENCE OF CLEAN FILTERS INSTALLED IN A PARALLEL-PIPED STANDBY FILTRATION PATH. TRANSFER IS READILY ACHIEVED BY OPENING TWO GATE VALVES. THERE ARE NO OTHER FILTERS IN THE INJECTION-FLOW PATH.

(5) EMPHASIS IS ON COMPARING PRESENT DESIGN WITH PROPOSED DESIGN

(6) ASSUMPTIONS TENDED TO PENALIZE PROPOSED DESIGN MUCH MORE THAN PRESENT DESIGN

(7) CRD SEAL LEAKAGE QUANTITY BASED ON ACTUAL DESIGN AND/OR TEST DATA

ATTACHMENT 2

ANALYSIS OF MAKE-UP CAPABILITY
SYSTEM CONDITIONS MODELED

GENERAL (ALL PRODUCT LINES)

(1) REACTOR PRESSURE AT RELIEF VALVE SAFETY SETPOINT + 15 PSI TOLERANCE + 20 PSI STATIC HEAD

(2) SINGLE PUMP AND DUAL PUMP OPERATION WITHOUT REGARD TO POSSIBLE SUCTION PRESSURE LIMITS

(3) PUMP DISCHARGE PRESSURE ASSUMED EQUAL TO PUMP TDH AT RATED FLOWS

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(5) EMPHASIS IS ON COMPARING PRESENT DESIGN WITH PROPOSED DESIGN

(6) ASSUMPTIONS TENDED TO PENALIZE PROPOSED DESIGN MUCH MORE THAN PRESENT DESIGN

(7) CRD SEAL LEAKAGE QUANTITY BASED ON ACTUAL DESIGN AND/OR TEST DATA

ALS:MKS/621

WITH RETURN LINE TO REACTOR VESSEL

BWR 4/5

(1) SCRAM VALVES OPEN

(2) PUMP TEST BYPASS OPEN

(3) MANUAL DISCHARGE PRESSURE CONTROL VALVE (PCV) OPEN (BWR/4's ONLY)

(4) SYSTEM FLOW CONTROL VALVE (FCV), DRIVE WATER PCV AND COOLING WATER PCV, REMOTE MANUAL AND BACKUP IN FULL OPEN POSITION

BWR/6

(1) SCRAM VALVES OPEN

(2) REMOTE SYSTEM FCV, DRIVE WATER PCV AND COOLING WATER PCV FULL OPEN

ATTACHMENT 3

BASIS FOR BOIL-OFF RATE ESTIMATES

(1) ANS JUNE 1978 BEST-ESTIMATE (I.E., +0%) DECAY HEAT CURVES

- INCLUDING: EFFECTS OF ACTINIDES

  DELAYED NEUTRONS

  2-YEAR AVERAGE CORE EXPOSURE

  EXPOSURE USED WAS THAT AT END OF FIRST CYCLE (BWR 5 & 6) AND AT END OF A RELOAD CYCLE (BWR 4). POWER HISTORY AT RATED POWER.

  ASSUMED CONSERVATIVE TWO YEAR AVERAGE EXPOSURE. "AVERAGE" MEANS CORE WIDE AND TIME AVERAGE.

  FINITE IRRADIATION EFFECTS ON U235/U238/Pu PROPORTIONS, HENCE ENERGY PER FISSION

  DECAY HEAT WAS CALCULATED BY TAKING THE ENERGY PER FISSION PER ISOTOPE TIMES ISOTOPIC COMPOSITION AT END OF CYCLE.

(2) FULL REACTOR POWER AT SCRAM

(3) NORMAL WATER LEVEL ABOVE TOP OF ACTIVE FUEL (TAF) WHEN CRD SYSTEM BECOMES SOLE HIGH PRESSURE WATER SOURCE-

(4) REACTOR WATER AT SATURATION TEMP AND PRESSURE (PRESSURE = RELIEF SET POINT +15 PSIG TOLERANCE)

ALS:RM/980
3/15/79

(5) 0.5 MWTH ENVIRONMENTAL LOSS INCLUDED

DESIGN BASIS FIGURE FOR NORMAL OPERATION AT RATED POWER HAS BEEN CALCULATED TO BE .6 MWTH. TO BE CONSERVATIVE WE PICKED .5 MWTH. THE IMPACT OF THIS FIGURE ON MAKE-UP FLOW IS INSIGNIFICANT.

(6) CRD/NUCLEAR BOILER LEAKAGE ACCOUNTED FOR BY ASSUMING THAT 5 GPM "UNACCOUNTED FOR" LEAKAGE IS LOST BY THE CRD INJECTION PIPING PRIOR TO ENTRY INTO THE RPV AND THAT THE REMAINING 25 GPM OF "ACCOUNTED FOR" NUCLEAR BOILER LEAKAGE OCCURS AFTER THAT WATER HAS BEEN HEATED TO SATURATION TEMPERATURE BUT BEFORE EVAPORATION TO STEAM TAKES PLACE (I.E., NO CREDIT TAKEN FOR LATENT HEAT OF EVAPORATION)

(7) VOLUME PER INCH OF REACTOR VESSEL ADJUSTED FOR SPECIFIC PLANT SIZE INTERNALS, LINEARIZED BETWEEN NWL AND TAF

(8) CRD WATER TEMPERATURE 80°F (NOMINAL)

(9) CRD SYSTEM BECOMES SOLE HIGH PRESSURE WATER SOURCE 40 MINUTES AFTER SCRAM COMPLETE

ATTACHMENT 4

BROWNS FERRY FIRE DOCUMENTED FACTS

FROM NRC REPORTS BASED ON TESTIMONY

OF CONTROL ROOM STAFF

	TIME	LAPSE
FIRE REPORTED	1235 HRS	
		16 MIN.
REACTOR SCRAM FROM 704 MWe POWER (MANUAL INITIATION)	1251 HRS	
		5 MIN.
ALL HP COOLING LOST EXCEPT CRD PUMP	1256 HRS	
		34 MIN.
REACTOR BLOWDOWN COMMENCED	1330 HRS	
		15 MIN.
CONDENSATE BOOSTER PUMP STARTED (1 OF 8 L.P. HIGH FLOW PUMPS) REACTOR PRESSURE BELOW 350 PSIG MIN. WATER LEVEL 48" ATAF REACHED	1345 HRS	
		12 MIN.
LEVEL STABILIZED AT 200" ATAF (i.e. NORMAL)	1357 HRS	
CRD FLOW DURING POST SCRAM PERIOD	100 GPM (AT FULL REACTOR PRESSURE)	
BOOSTER PUMP FLOW AT 350 psi	10,000 GPM TOTAL (BULK BYPASSED TO PUMP SUCTION)	
REMOTE MANUAL CONTROL OF RELIEF VALVES LOST DUE TO MALFUNCTION OF CONTROL AIR SOLENOID VALVES. LEVEL AT 217" ATAF. PRESSURE AT 200 PSIG AND RISING	1810 HRS	
		~50 MIN.
4" MAIN STEAM LINE DRAIN OPENED VENTING SOME STEAM TO SUPPRESSION POOL. PRESSURE 420 PSIG AND RISING	1900 HRS	
		~150 MIN.
RELIEF VALVE REMOTE MANUAL CONTROL RESTORED. BLOWDOWN COMMENCED (2 RV's) PEAK PRESSURE RECORDED 600 PSIG	2130 HRS	
		50 MIN.
CONDENSATE BOOSTER PUMPS CONTROLLING REACTOR LEVEL. PRESSURE 220 PSIG AND FALLING.	2220 HRS	

Updated 8/30/78
E.J. Penn

mh/1596

ATTACHMENT 5

BROWNS FERRY 1 AND THE IMPACT OF THE PROPOSED DESIGN

WOULD THE PROPOSED CHANGE HAVE AFFECTED THE OUTCOME OF THE BF 1 FIRE?

NO

THE MINIMUM REACTOR WATER INVENTORY WOULD HAVE BEEN UNCHANGED HAD RETURN LINE BEEN CUT AND CAPPED (WITHOUT REROUTE) PRIOR TO THE BF 1 FIRE BECAUSE THE PROPOSED CHANGE AND PRESENT DESIGN WITHOUT VALVE ADJUSTMENT HAVE IDENTICAL FLOW RESISTANCE IN THE SCRAM MODE WHICH SET THE CONTROLLING FLOW PATH. BOTH THE OLD SYSTEM AND NEW SYSTEM HAVE IDENTICAL FLOWS WITH THEIR VALVES IN THEIR RESPECTIVE NORMAL POSITIONS.

WOULD THE OUTCOME OF THE BF 1 FIRE HAVE BEEN DIFFERENT IF THE CRD PUMP HAD FAILED DURING SCRAM AND REMAINED INOPERATIVE THEREAFTER?

YES

ESTIMATED LOWEST INVENTORY DURING THE SECOND REACTOR BLOWDOWN (2130 TO 2220 HRS) WITH CRD FLOW.	105" ATAF*
ESTIMATED CONTRIBUTION OF CRD SYSTEM (VIA DRIVES) DURING PERIOD (1810 TO 2220 HRS).	195"
FINAL INVENTORY HAD CRD PUMP FAILED	90" BTAF**

* ABOVE TOP OF ACTIVE FUEL
**BELOW TOP OF ACTIVE FUEL

ALS:BP/1053
10/16/78

ANALYSIS OF CONTRIBUTION OF THE CRD SYSTEM DURING BF1 FIRE

1. Water volume per inch of vessel level (at 60°F) = 193 gal/inch

2. Reactor water level at scram (1251 HRS) = 201" ATAF

3. Total time for which CRD system was sole source 1256 hrs. to 1330 hrs. prior to completion of first reactor blowdown = 34 minutes

4. Contribution of CRD system while sole source = 34 x 100 gallons = $\frac{3400}{193}(\text{inch}) \times \frac{62.22}{53.65}$ (density ratio) = 20.4 inches

5. Contribution of CRD system between scram and becoming sole source = 5 x 100 = 500 gallons = $\frac{500}{193}\text{ inch} \times \frac{62.22}{53.65}$ = 3.0 inches

6. Level at end of first blowdown, at which time 8 alternates to the CRD pump become available = 48 inches ATAF

7. Level at end of first blowdown if CRD pump had FAILED following scram = 48 - 3.0 - 20.4 = 24.6 inches ATAF

8. Level at end of first blowdown if max. single pump capability had been used with return line
$= 39\left(\frac{180 - 100}{193}\right)\left(\frac{66.22}{53.65}\right) + 48$ = 66.7 inches ATAF

9. Level at end of first blowdown if max..single pump capability of proposed system (no return line) had been used.
$= 39\left(\frac{170 - 100}{193}\right)\left(\frac{62.22}{53.65}\right)$ = 64.4 inches ATAF

10. Level at end of first blowdown if CRD return line had been previously cut and capped at BF1. (With one pump).
$= 48 - \frac{(39 \times 5)}{193}\left(\frac{62.22}{53.65}\right)$ = 46.8 inches ATAF

11. Latest time to start first blowdown without uncovering TAF:

48 x 193 = 9264 gallons of remaining inventory would boil-off in approximately 30 minutes (boil-off rate analysis model). Blowdown started 1330 hrs.

Latest blowdown decision to keep core covered 1330 + 30 1400 HRS

12. Estimated core uncovery time if first blowdown not initiated (GE boil-off rate model) 1440 HRS (109 m after scram)

ANALYSIS (CONTINUED)

13. Estimated contribution of CRD flow between 1810 and 2130 hrs. (i.e., start of second blowdown) = 156

= 130 gpm (average) x 200 min.

$$\left(\frac{26000}{193}\right)\left(\frac{62.22}{53.65}\right) = 156.2$$

14. Estimated contribution of CRD during blowdown

= 130 gpm (average) x 50 min.

$$\left(\frac{6500}{193}\right)\left(\frac{62.22}{53.65}\right) = 39.4$$ = 39"

TOTAL = 195"

15. Estimated water level at time 2130 hr. without CRD flow (Condensate booster pump flow not adjusted to compensate)* assuming level of 217" ATAF at 1810 hrs. = 40" ATAF

16. Estimated level loss during second blowdown (due to steam generated by decay ht. and release of stored heat in water inventory) = 130"

17. Estimated final level at 2220 hrs. without CRD flow = 90" BELOW TAF

18. Estimated final level with CRD flow

= -90 + 195 = 105 = 105" ABOVE TAF

*In the event that the CRD pump had become unavailable following scram the condensate booster pump injection flow would have been increased to compensate. The estimated effect of such added booster pump flow would have been to increase the level at 2130 hrs. to 65" ATAF and at time 2220 hrs. to 80" below TAF.

Updated 8/30/78
E.J. Penn

mh/1597-1598

GENERAL ELECTRIC

NUCLEAR ENERGY PROJECTS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

April 9, 1979

MFN 100-79

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. V. Stello, Jr., Director
Division of Operating Reactors

Mr. R. J. Mattson, Director
Division of Systems Safety

Gentlemen:

SUBJECT: SOLENOID VALVE ENDURANCE TEST - NEDE 24611

Reference: March 14, 1979 letter, G. G. Sherwood (GE) to V. Stello (NRC) and R. J. Mattson (NRC), subject: Control Rod Drive (CRD) Return Line Removal

The subject report is submitted to assist in the evaluation of the GE proposal to delete the CRD return line as discussed in the referenced letter. This report is forwarded in response to the request of Mr. R. P. Snaider of the NRC staff.

If you require additional information, please contact Alvin L. Spivak of my staff at (408) 925-3875.

Very truly yours,

Glenn Sherwood

Glenn G. Sherwood, Manager
Safety and Licensing Operation

GGS:bjw/1166

Enclosure

cc: Fred Clemenson, NRC, w/o
L. S. Gifford, GE-Bethesda w/o

GENERAL ELECTRIC

NUCLEAR ENERGY

PROJECTS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

MFN 110-79

May 1, 1979

RECEIVED
MAY 04 1979
A.J. LEVINE

bcc:
RO BRUGGE
D FISCHER
RL GRIDLEY
AJ LEVINE
MA ROSS
AL SPIVAK
EP STROUPE
DG TIBBILS
HT WATANABE

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. V. Stello, Jr., Director
Division of Operating Reactors

Mr. R. J. Mattson, Director
Division of Systems Safety

Gentlemen:

SUBJECT: TEST RESULTS FOR SOLENOID VALVE PERFORMANCE TEST - NEDE 23887

Reference: March 14, 1979 letter, G. G. Sherwood (GE) to V. Stello (NRC) and R. J. Mattson (NRC), subject: Control Rod Drive (CRD) Return Line Removal

The subject report is submitted to assist in the evaluation of the GE proposal to delete the CRD return line as discussed in the referenced letter. This report is forwarded in response to the request of Mr. Fred Clemenson of the NRC staff.

If you require additional information, please contact Alvin L. Spivak of my staff at (408) 925-3875.

Very truly yours,

Glenn G. Sherwood

Glenn G. Sherwood, Manager
Safety and Licensing Operation

ALS
Ray - one more to go!
Thanks
ajl
5/4

GGS:hcl/1234

Enclosure

cc: R. P. Snaider, NRC, w/att subject
L. S. Gifford, GE-Bethesda w/o att
File:

bcc: (w/o) Enclosures
E. P. Stroupe
R. O. Brugge
Dieter Fischer
A. J. Levine
H. T. Watanabe
M. A. Ross with Encl.

GGS:pes/812

ATTACHMENT NO. 1

GENERAL ELECTRIC REPORT

"EVALUATION OF THE EFFECTS OF CORROSION PARTICLES

ON CONTROL ROD DRIVE OPERATION"

EVALUATION OF THE EFFECTS OF CORROSION PARTICLES ON CONTROL ROD DRIVE OPERATION

1.0 INTRODUCTION

In late 1976, cracks were observed in the vicinity of the Control Rod Drive (CRD) return line nozzle of several BWR reactors. The cause of the cracking was determined to be thermal cycling of the nozzle region due to the relatively cool CRD hydraulic system return line flow. In order to avoid thermal cycling and the possible cracking of the CRD return line nozzle, the return line was either isolated or removed.

With the CRD return line isolated or removed, the CRD system flow is changed such that there may now be flow paths to the drives which pass through runs of carbon steel pipe without subsequent filtering. The NRC has expressed concern (References 1 and 2) that operation under these conditions would increase the likelihood of foreign material being introduced into the drives in such a fashion and quantity as to potentially impair their ability to properly respond to a scram signal.

This report addresses the potential impact of foreign material being introduced into the drives in such a manner as to potentially impair their ability to properly respond to a scram signal.

2.0 SUMMARY

This report evaluates the effects of corrosion particles, assumed to be generated in the carbon steel piping of the flow stabilizer loop and the exhaust water header, on the operation of the CRD system. The results of the evaluations indicate that the presence of corrosion particles does not affect the reliability of the scram function of the CRD system.

The possible effects of corrosion particles from the flow stabilizer loop on the operation of the drive piston, the cooling water orifice, and the ball check valve are analyzed. The possible effects of corrosion particles from the exhaust water header on the operation of the No. 121 directional control valves and associated filters are also analyzed.

3.0 ANALYSIS

3.1 CRD System Flow Changes

When the CRD hydraulic system is operated with the CRD return line isolated or removed, the CRD system flow is changed. The system flow changes result in two new flow paths to the drives which pass through runs of carbon steel piping. As shown in Figure 1, these flows are:

1. The flow through the stabilizer loop, which then goes up the cooling water header to the bottom of the drives, and
2. The reverse flow up through the exhaust water header, through the No. 121 directional control solenoid valve, and then up to the top of the drives.

The possible effects of corrosion particles from the flow stabilizer loop on the operation of the drive seals and cylinders, the cooling water orifice, and the ball-check valve are discussed in Section 3.2.

The possible effects of corrosion particles from the exhaust water header on the operation of the 121 directional control valve, and the adjacent filter No. 136, are discussed in Section 3.3.

3.2 Flow Stabilizer Loop Corrosion Particles

With the CRD return line isolated or removed, flow from the flow stabilizer loop is discharged to the reactor pressure vessel (RPV) via the drive cooling water flowpath. Since the piping downstream of the stabilizer valves is carbon steel, the flow passes through about 12 feet of carbon steel piping before it joins the cooling water flow.

The stabilizer loop has a continuous flow of 2 to 5 gpm, at relatively low temperatures. Therefore, any carbon steel corrosion particulate matter which enters the flow will be small in quantity and particle size. The limiting condition is a situation in which relatively large sized particles are spalled from the carbon steel piping at a high rate. Assuming that these particles remain entrained in the cooling water header flow, (i.e., the particles do not settle out at low points in the piping), the particles could eventually accumulate in the control rod drives, since the cooling water flow path to the drive is essentially free of flow restrictions and active valves.

Once inside the drives, the corrosion particles could go: (1) into the underside of the drive piston; (2) up through the cooling water orifice; or (3) settle about the scram water ball-check valve. Each of these is considered in subsections following.

3.2.1 Corrosion Particles Below the Drive Piston

If the corrosion particles were to enter the volume below the drive piston, their presence would go unnoticed. During periods of normal plant operation, they would settle to the bottom of the drive, away from the drive piston above. These particles would contribute little to the scoring of cylinder walls, or to seal failures.

Since the pressure and flow to the bottom of the drive is highest during scram operation, the potential for lifting the corrosion particles in suspension and forcing them into the drive seals is the greatest during scram. However, even during scram operations, the corrosion particles would not cause significant scoring of the cylinder walls, or seal failures, such that the operation of the drive would be affected.

However, abrasive corrosion particles could accelerate wear of the drive cylinders and seals. The minimum performance requirements of the drives during reactor operation is specified in

the Technical Specification. If the limiting conditions for operation are not met, the CRD is considered inoperative and the subsequent operation of the reactor is adjusted, as required, to account for the inoperative drive.

The drive water pressure and flow available for rod position changes is less than that delivered to the drive during a scram. Therefore, the condition of the drive components is much more essential to normal rod positioning than to scram operation. Consequently, the drives will continue to meet scram performance criteria even after they have failed to meet the limits for normal rod movement and have, subsequently, been identified as inoperative drives. Hence, scram actions will, in no manner, be inhibited.

3.2.2 Corrosion Particles in the Cooling Water Orifice

It is also possible that corrosion particles from the carbon steel piping in the flow stabilizer loop may be carried up toward the cooling water orifice, where some particles may eventually become lodged in the orifice itself. Theoretically, the orifice could become totally plugged.

Although this is not desirable, it is not a safety problem. Blocking the cooling water orifice would cause the drive to heat up. Continuing plant operation with such a "hot-drive" would produce a more rapid degradation of the drive piston seals and bushings. Installation of a particulate filter (50 micron) in the cooling water header would eliminate this as a concern.

An increase in the rate of seal and bushing deterioration, as mentioned in the case of corrosion particles in the drives themselves, will first be noted during normal rod operation.

3.2.3 Corrosion Particles About the Ball-Check Valve

If corrosion particles were to settle about the scram ball-check valve, their presence would not be expected to affect the operation of the drive. There is sufficient clearance around the check ball and its cage so that corrosion particles would offer little resistance relative to the large upward force exerted on the ball following a scram at elevated reactor vessel pressure. At low vessel pressures, the check valve is not required to open to complete the scram.

The corrosion particles would settle about the ball-check valve, and pile-up like silt. Corrosion particles could get under the check ball (between the check ball and the valve seat), and prevent the ball from seating properly. However, although operational problems may be encountered during normal rod position changes (reduced insert and increased withdrawal speeds), no safety problems would result, because the pressures and flows delivered to the drive during a rod scram are sufficient to compensate for possible leakage around the unseated ball-check valve.

3.3 Exhaust Water Header Corrosion Particles

The elimination or isolation of the return line results in a reverse flow through the No. 121 directional control valves (see Figure 1). This reverse flow comes from two sources: 1) the reverse flow up through the exhaust water header orificed check valve (No. 100); and 2) the exhaust water flow discharged from adjacent drives.

Considering the limiting condition of relatively large sized corrosion particles spalled from the exhaust water header at a high rate, the potential consequences may be operational problems, but not safety problems. Particles carried by the reverse flow from the exhaust water header up through the 121

3.2.3 Corrosion Particles About the Ball-Check Valve

If corrosion particles were to settle about the scram ball-check valve, their presence would not be expected to affect the operation of the drive. There is sufficient clearance around the check ball and its cage so that corrosion particles would offer little resistance relative to the large upward force exerted on the ball following a scram at elevated reactor vessel pressure. At low vessel pressures, the check valve is not required to open to complete the scram.

The corrosion particles would settle about the ball-check valve, and pile-up like silt. Corrosion particles could get under the check ball (between the check ball and the valve seat), and prevent the ball from seating properly. However, although operational problems may be encountered during normal rod position changes, no safety problems would result, because the pressures and flows delivered to the drive during a rod scram are sufficient to compensate for possible leakage around the unseated ball-check valve.

3.3 Exhaust Water Header Corrosion Particles

The elimination or isolation of the return line results in a reverse flow through the No. 121 directional control valves (see Figure 1). This reverse flow comes from two sources: 1) the reverse flow up through the exhaust water header orificed check valve (No. 100); and 2) the exhaust water flow discharged from adjacent drives.

Considering the limiting condition of relatively large sized corrosion particles spalled from the exhaust water header at a high rate, the potential consequences may be operational problems, but not safety problems. Particles carried by the reverse flow from the exhaust water header up through the 121

directional control valves will affect only two components: the 121 valve itself, and the filter element No. 136 just downstream of the 121 valve in the reverse flow direction.

3.3.1 Corrosion Particles in the Directional Control Valve Filter

Filter element No. 136 is a 50 micron filter, and it will trap the large corrosion particles which have the greatest potential for causing scoring or accelerated wear in the drives. If the filter became plugged with corrosion particles, flow through the filter in both directions could be restricted. The flow restriction presented by the plugged filter could prevent adequate venting of the fluid above the drive piston and restrict upward rod movement when the rod is to be inserted one or more notches.

However, the plugging of filter 136 would not affect scram performance in any way. Venting of the fluid above the top of the drive piston during a scram is accomplished by the opening of the scram discharge valve located upstream of the 136 filter.

If the filter became plugged to the extent that the limiting condition for drive operation in the Technical Specifications could not be met, maintenance would be performed on the Hydraulic Control Unit (HCU) and the filter would be replaced. If such corrective action were not taken, the drive would be identified as an inoperative drive and removed from service.

3.3.2 Corrosion Particles in the Directional Control Valve

The limiting case would be one in which the corrosion particles could theoretically cause the 121 valves to stick in the wide-open or fully-closed position.

If the 121 valves were to stick wide-open, problems would be encountered when attempting to withdraw, or "notch out" the rod. A portion of the drive water flow to the top of the drive would be discharged to the exhaust water header through the open 121 valve.

If the 121 valves were to stick closed, the consequences would be the same as those previously discussed for a plugged 136 filter, (i.e., problems during rod insertion, or "notch in" operation).

Neither of these two situations would prevent the safe shutdown of the reactor (Reference 3). Failure of the 121 valves in the wide-open or fully-closed position affects only normal rod position change operations. If not corrected, failure of the valve in either position would result in the drive being designated as inoperative.

4.0 CONCLUSIONS

This evaluation of the effects of carbon steel corrosion particles on the operation of the CRD system with the return line isolated supports the following conclusions:

1. The potential presence of corrosion particles will not affect the scram function of the Control Rod Drive System.

2. Long-term operation of the CRD system in its present configuration may accelerate operational and maintenance problems unless appropriate corrective action is taken to replace the subject carbon steel piping, or to filter the flows that run through it. However, field experience totaling about 30 reactor years of operation over the past 2 years has not identified operational problems. Because there has been no reported adverse experience, the evaluation reported herein is conservative, and no performance trend changes are anticipated.

5.0 REFERENCES

1. Amendment No. 30 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant, dated September 16, 1977.

2. Letter, G. Lear (NRC) to PASNY, "Safety Evaluation of Proposed Reduction in Frequency of Control Rod Scram Surveillance Requirements, James A. FitzPatrick Nuclear Power Plant, Docket No. 50-333," dated June 1, 1978.

3. "BWR Scram System Reliability Analysis," NEDE-21514, dated December, 1976.

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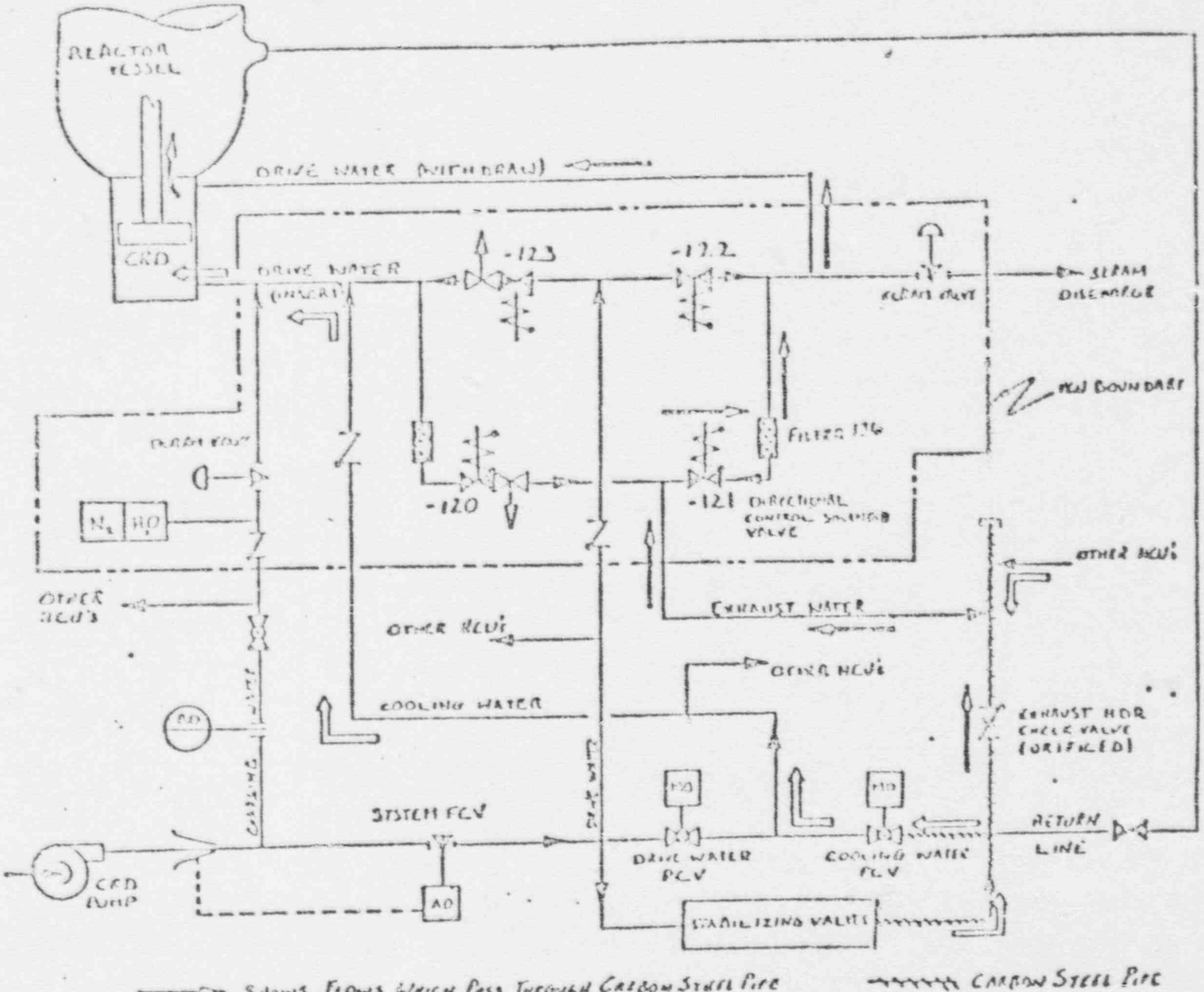


Figure 1. CRD System Flows Through Carbon Steel Piping

ATTACHMENT NO. 2

GENERAL ELECTRIC REPORT

"CRD HYDRAULIC CONTROL SYSTEM

RETURN LINE MODIFICATION"

April 1979

CHANGES TO THE RETURN LINE TO THE REACTOR VESSEL
CRD HYDRAULIC CONTROL SYSTEM

Intergranular stress corrosion cracking in the Control Rod Drive (CRD) hydraulic control system return line near its connection to the Reactor Pressure Vessel (RPV), and fatigue cracking in the reactor vessel nozzle blend radius, have occurred at several operating BWRs. The stress corrosion cracking has occurred in Type 304 stainless steel piping and in the RPV nozzle safe ends at weld-heat-affected zones. The fatigue cracking in the RPV nozzles is caused by thermal cycling of the relatively cold water in the return line during different modes of CRD operation. This memorandum is a review of operating experience relative to this cracking, and a consideration of alternative corrective measures.

CONCERNS

Crack growth in the CRD return line and RPV nozzle can lead to the need for line replacement, vessel repair, and loss of plant availability. Therefore, inspection and crack removal, if required, generally has been performed at the earliest outage opportunity. To perform an adequate inspection, thermal sleeves are fully or partially removed from nozzles that are equipped with them. The sleeve is not reinstalled if the CRD return line nozzle is capped. If the return line nozzle is not capped, the thermal sleeve or a modified thermal sleeve is reinstalled even if the return line is valved closed with only infrequent use of the return flow capability being planned.

There is a licensing concern regarding changes in the reactor vessel water makeup capability from the CRD hydraulic control system. As such, the makeup capabilities both before and after a proposed modification should be discussed in the licensing submittal. In the case of a re-routing, there would be minimal change in make-up capability.

ALS:at:rm/58A

CRD HYDRAULIC CONTROL SYSTEM

Figure 1 shows a typical CRD hydraulic control system with a return line. In this original design the CRD system operation was such that the flow in the return line to the RPV was the excess CRD system flow (i.e., the portion of the total CRD system flow which had not been used for accumulator charging, CRD cooling, or CRD operation). The function of the return line was to provide excess flow to stabilize the control system and to return exhaust water from operating CRDs to the RPV.

PERFORMANCE WITHOUT THE RETURN LINE

When the return line valve is closed or the line is removed, all system flow is through the CRDs. In this configuration it is necessary to readjust the control valves to provide proper flows and pressures in the system. When this is done a constant one to two gallons per minute reverse flow from the CRD pump discharge passes through the orificed, exhaust header check valve (V100) due to increased pressure in the valved closed return line. The exhaust header pressure also increases until it is sufficient to cycle one or more of the insert exhaust directional solenoid valve pistons (Valves No. 121) allowing the one to two gallons per minute to continuously backflow through the valves into the vessel. The major considerations with this operating mode are:

CRD Performance

CRD movement may be somewhat slower since the exhaust header pressure is higher. It should be possible to compensate for any loss of rod speed by normal rod speed adjustments. Consequently, it is important to perform an isolated operation test giving careful attention to drive settling performance. In-plant testing has confirmed satisfactory operation of the CRDs. The system is stable without the return line while the exhaust water from operating CRDs is returned by backflow through the 121 valves (with the one to two gpm mentioned earlier).

Effects of Reverse Flow on CRD Components

The original system design allowed for some cycling of the 121 valves to occur during drive motion and recent tests of ASCO solenoid valves (1.5 million cycles per valve) have shown no loss of valve function. Consequently, temporary operation of the CRD system in an isolated mode is acceptable for BWRs 2, 3, 4 and 5.

Removal of Exhaust Header Connection to CRD Pump Discharge

To limit reverse flow cycling of the 121 valves to periods of CRD operation and allow permanent CRD operation without a return line or with the return line valve closed, removal of the exhaust header connection to the CRD pump discharge is necessary as shown in Figure 2. This modification eliminates the continuous backflow through V100 and involves the following:

1. Cooling Water Pressure Control Valve - Remove or open permanently.

2. Equalizing Valves (Valve Number F150) - Add two valves as shown in Figure 2 between the cooling water line and the exhaust header to re-pressurize the exhaust header following a scram and prevent excessively high CRD operating differential pressure during subsequent operation of a selected CRD.

3. Exhaust Header Check Valve - Remove and cap exhaust header.

PIPE MATERIAL EFFECTS

With the CRD return line isolated or eliminated, the CRD system flow is changed such that there may be flows to the CRDs which pass through runs of carbon steel pipe without subsequent filtering. As shown in Figure 1, these runs of carbon steel pipe are in the exhaust water header and in the flow stabilizer loop. To assure that the quality of the water delivered to the CRDs is not degraded by the removal of the return line flow, appropriate actions should be taken to eliminate or control the potential carbon steel corrosion products which may flow toward the

drives. Recommended actions are to replace all carbon steel piping with stainless steel, or, install particulate filter (50 micron) in the cooling water header to protect cooling water orifice and establish procedures to flush the exhaust water header periodically.

OTHER VARIATIONS

Operation With Existing Return Line Valve Closed

Operation in an isolated mode (with the existing CRD return line valve closed) is an appropriate interim action until other modifications can be made. This will decrease the thermal fatigue cracking potential of the nozzle but not necessarily the stress corrosion cracking potential of the piping and safe end (since the line is subjected to an environment that is conducive to stress corrosion cracking). Also, operation in this mode results in continuous cycling duty on the 121 valves unless the exhaust header arrangement is modified as in Figure 2. Ten units are reported to be operating in this mode.

Remove the Return Line, and Cap the Vessel Nozzle

Removal of the return line, capping the vessel nozzle and isolating the exhaust header from the CRD pump discharge as shown in Figure 2 are acceptable as long as the CRDs operate satisfactorily and the makeup capability is acceptable. In-service inspection requirements are reduced, and cycling duty on the 121 valves is limited to periods of CRD operation. One unit is reported to be operating in this mode.

Replace the Return Line, Modify the Exhaust Header, and operate with the Return Line Valve Closed.

Replacing the return line with nuclear grade 316L stainless steel will reduce the stress corrosion cracking potential; and operation with the return line valve closed and the exhaust header modified as in Figure 2 will resolve the vessel nozzle thermal fatigue cracking problem. CRD performance is the same as above, but in-service inspection is greater.

Reroute the Return Line, Cap the Vessel Nozzle, and Either Operate with the Return Line In-Service, or Valve Closed

Rerouting the return line to another system, such as the feedwater or cleanup system, may result in a cracking problem in that system unless material selection and piping design are such that the risk of stress corrosion cracking and thermal fatigue is minimized. The connection point to the other system must be such that the effects of system fluctuations on the exhaust header pressure do not impact CRD system operation. In this respect, operation of the CRD system will be simplified by operating with the reroute line isolated and the exhaust header arrangement modified as in Figure 2. The return line can be rerouted to the cleanup system and operated with the line valved closed. Seven units are reported to be operating in this mode.*

IN SUMMARY

1) Control Rod Drive performance without return line flow is satisfactory.

2) For long-term operation with the return line isolated or with the return line removed, it is necessary to isolate the exhaust header from the CRD pump discharge as in Figure 2.

3) Operation with the present return line valved closed is only an interim solution to the cracking problem, since the line is still susceptible to stress corrosion.

RECOMMENDATIONS

The General Electric recommendations (for BWRs 2, 3, 4 and 5) to remove the potential for cracking of the CRD return line and its vessel nozzle are as follows:

ALS:at:rm/58A

a) At the first shutdown opportunity, perform a test with the return line valved closed and the control valves readjusted to verify that CRD operation is acceptable and then operate on an interim basis with the return line valved closed.

b) During the next convenient outage, inspect the return line and vessel nozzle and remove any cracks.

c) At the first convenient outage following licensing approvals:

   1. Modify the exhaust header arrangement as in Figure 2 (use stainless steel piping or install filters and periodically flush the exhaust header).

   2. Permanently remove the CRD return nozzle thermal sleeve, if any.

   3. Examine the CRD return nozzle bore, blend radius and vessel wall adjacent to the nozzle. Remove crack indications, if any.

   4. Remove the CRD return line piping and safe end and install a cap on the CRD return nozzle. For BWRs in which the CRD return line is optionally rerouted, operate the CRD system with the rerouted return line valve closed.

SIMPLIFIED SCHEMATIC
TYPICAL CRD HYDRAULIC CONTROL SYSTEM

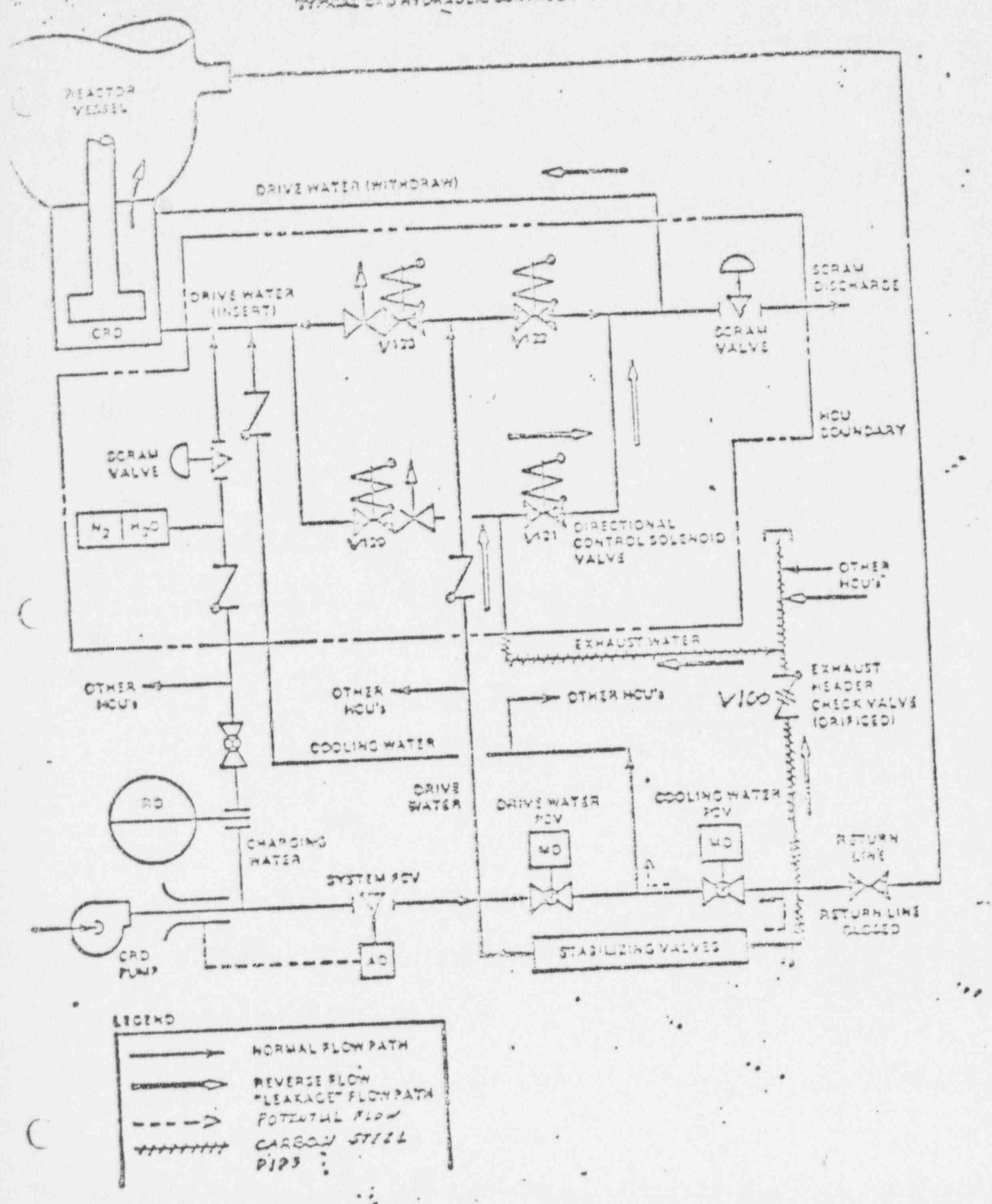
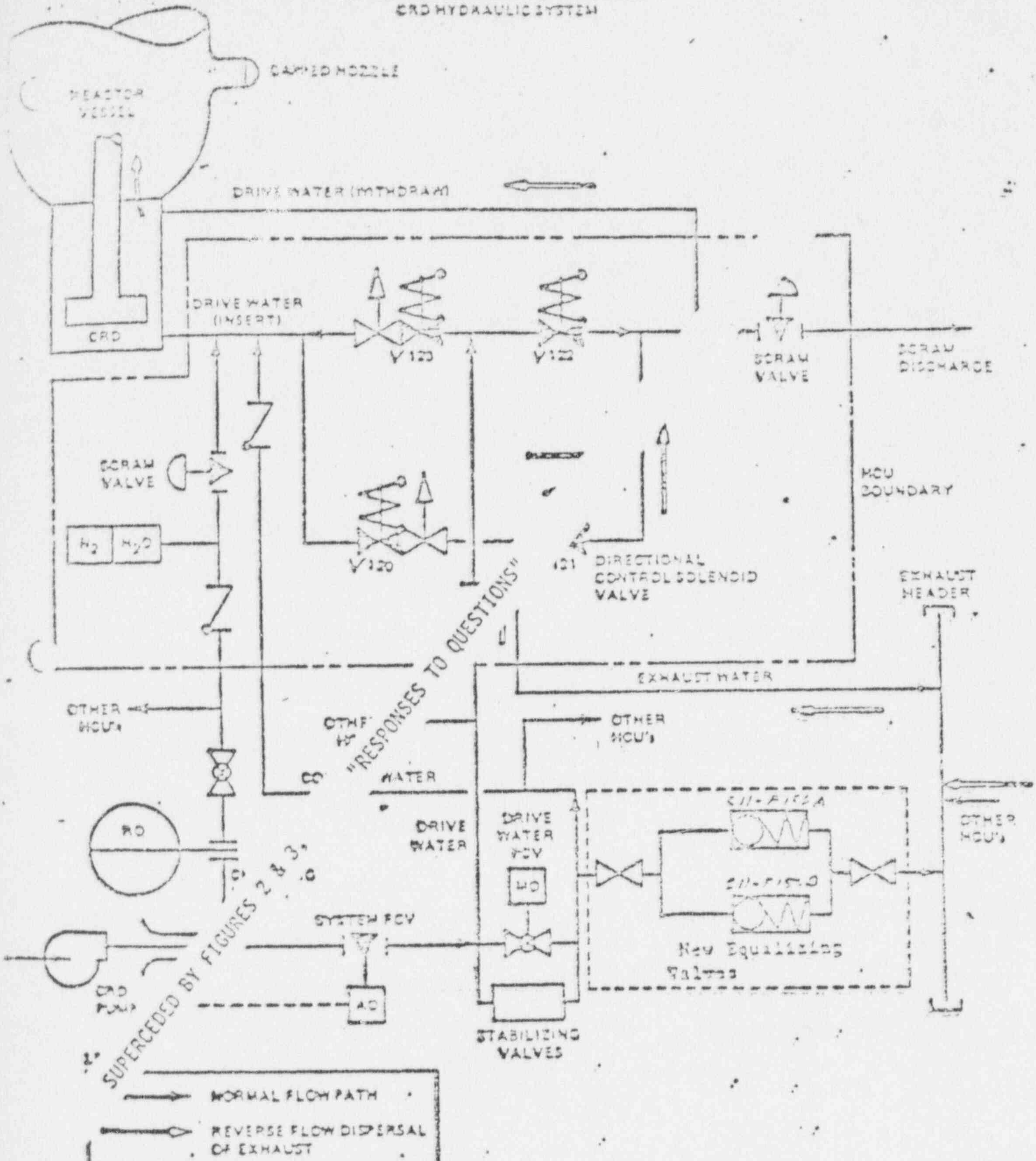


FIGURE 2

SIMPLIFIED SCHEMATIC OF MODIFIED CRD HYDRAULIC SYSTEM



ATTACHMENT NO. 3

GENERAL ELECTRIC

RESPONSES TO NRC CONCERNS

REGARDING

CORROSION PRODUCT GENERATION

ALS:at/92J3

REQUEST FOR ADDITIONAL INFORMATION
GENERIC REVIEW OF BWR CRD MODIFICATIONS

1. In reference to the May 22, 1979 submittal on CRD Hydraulic Control System Return Line Modifications for Operating Plants, provide the following:

   a. Should a licensee choose to retain the existing carbon steel piping rather than replacing it with stainless steel, describe and discuss the circumstances under which it would be acceptable to permit power operation without installing the 50 micron particulate filters in the cooling water line.

RESPONSE

The original GE recommendation for the installation of the subject filters in the cooling water header was primarily based on the expectation that, with the return line eliminated, there would be flow from the exhaust water header to the cooling water header during periods of control rod movement. The exhaust water header is typically a 200 foot length of 1" or 1½" carbon steel pipe. Due to this relatively long length of carbon steel piping, it was the judgement of GE Engineering that the subject filters should be added to filter the flow from the exhaust water header.

Subsequent testing at an operating plant, however, revealed that the flow to the exhaust water header from drive movements was not discharged into the cooling water header. Instead, the flow exhausted from the moving drive was observed to discharge directly back up to the reactor pressure vessel (RPV) via a reverse flow path through the insert exhaust directional control valves, (i.e., the HCU-121 valves), of adjacent Hydraulic Control Units (HCUs). With the recommended installation of the pressure equalizing valves between the cooling water and exhaust

water headers, the stabilizer loop flow is consequently the sole source of flow to the cooling water header which goes through carbon steel pipe and is not subsequently filtered.

The potential for carbon steel corrosion particles to be generated in this short run of piping is addressed in detail in the response to question #3. As concluded in that response, the contribution of carbon steel corrosion particles to the cooling water flow from the carbon steel piping of the stabilizer loop is comparatively small in relation to the carbon steel corrosion particles which are already in the flow. Therefore, although the GE recommendation is still directed toward a more absolute solution through the replacement of the carbon steel piping in the flow stabilizer loop with stainless steel or the installation of the subject 50μ filters in the cooling water header, the alternative action of "do nothing" is considered to be acceptable.

AS:cas/92G2

b. Assuming the above mentioned 50 micron filters are installed, describe and discuss the recommended method and frequency of cleaning the carbon steel pipes.

RESPONSE

Cleaning of the carbon steel piping is essentially the flushing of the pipes of the loose carbon steel corrosion particles which may have accumulated. This flushing operation may be accomplished by simply running a high volume flow through the line or possibly through the use of a hydrolaser.

The frequency at which such flushing is required is going to depend on how fast the crud accumulates in the pipe. The rate of crud accumulation will vary primarily as a function of the quality and oxygen content of the water and is, therefore, highly plant dependent and difficult to define. The best indicator of the need for flushing the exhaust water header piping is probably dirty filters in the HCUs. An estimation of the required frequency for flushing the exhaust water header is somewhere between 5 and 10 years of operation. As for the carbon steel in the flow stabilizer loop, it is continually being flushed and should require no additional maintenance flushing.

AS:cas/92G3

c. Explain why the 50 micron particulate filters are not shown on Figure 2 of the modified CRD hydraulic system.

RESPONSE

The GE recommendation is to eliminate the carbon steel in the flow stabilizer loop. With this run of carbon steel pipe eliminated, there is no flow downstream of the main drivewater filters which passes through carbon steel and thus, no need to filter the cooling water flow. Therefore, the referenced figure depicts the flow stabilizer loop rerouted directly to the cooling water header and all carbon steel eliminated from this flow path.

AS:cas/9204

d. Provide the circumstances or reasons, implied in the May 22, 1979 cover letter, why a utility could consider making modifications to the CRD hydraulic system without submitting a request to NRC. The response should address the staff's concern that corrosion products may collect between the check ball and its cage to such an extent as to jam the ball.

RESPONSE

Potential for jamming of check ball is discussed in the response to question No. 3.

AS:cas/92G5

e. The need for CRD system modifications was initially addressed in Service Information Letter (SIL) 200 dated October 29, 1976. Considering that the concern is not fully resolved to date, quantify and supply the supporting information for the term "interim" in the following sentence: "Operation with the present return line closed is only an interim solution to the cracking problem, since the line is still susceptible to stress corrosion."

RESPONSE

As stated in the referenced original draft of SIL 200 and in subsequent supplements to SIL 200, cracks in the RPV nozzle for the CRD Hydraulic System return line were found in six (6) plants. Evaluation of these cracks revealed evidence of both intergranular stress corrosion cracking (IGSCC) and fatigue cracking. Factors contributing to these types of cracks were the thermal gradients and thermal cycling resulting from the injection of the cool CRD system flow.

To preclude further nozzle cracking it was deemed most prudent to eliminate the source of these thermal mechanisms which in this case was the CRD system return line flow. Appreciating that a final fix to the problem would take time to implement (irrespective of whether the final solution was rerouting or eliminating the return line), the interim solution of isolating the return line was offered. Although isolating the return line eliminates the crack-causing contributions of thermal gradients and thermal fatigue, it does not fully resolve the problem of IGSCC. Therefore, isolation of the return line is considered to be, as stated, only an interim solution. For further information concerning IGSCC, please refer to the July 1975 submittal to the NRC on this subject.

AS:cas/92G6

Furthermore, to answer the question "what precautionary measures have been taken to address the continued concern of IGSCC in the isolated CRD system return line", GE has encouraged BWR owners to follow the recommendations of the NRC Pipe Crack Study Group as documented in Report NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants". In the NUREG-75/067 report, the NRC Pipe Crack Study Group has recommended: (1) certain austenitic stainless steel type 304 and 316 lines be replaced with materials less susceptible to stress corrosion cracking; or if not replaced, (2) an augmented ISI program should be enacted with increased examination frequency. Thus, the GE recommendation for isolated CRD system return lines is to increase the surveillance of the portion of the return line which is subject to regular Inservice Inspection.

AS:cas/92G7

2. Item 3 of Attachment C to the January 27, 1978 submittal indicates that the recommended equalizing valves prevent a high pressure differential between the cooling water header and exhaust header. Further, this potentially high pressure differential may result in an excessive initial control rod withdrawal speed. Describe and discuss the following:

   a. The maximum potential magnitude of this pressure differential.

RESPONSE

The purpose of the subject pressure equalizing valves is to assure that the pressure in the exhaust water header is at essentially the same pressure as the RPV. The equalizing valves are positioned between the cooling water header and the exhaust water header. Since the cooling water header is maintained at approximately 20 psi above the referenced vessel pressure and the equalizing valves open at approximately 80 psid, this arrangement assures that the exhaust water header is maintained within about 60 psi of the referenced vessel pressure.

The equalizing valves are installed in a redundant parallel configuration. In the event one of the valves should fail, the second (i.e., redundant) valve will perform the intended function of maintaining the exhaust water header near the pressure of the vessel. In the unlikely situation that both valves failed to function, the differential pressure between the RPV and the exhaust header could be as great as 1050 psi.

AS:cas/92G8

b. The various modes of reactor operation that lead to high pressure differentials, and the estimated frequency of their occurrence.

RESPONSE

Assuming that there is no pressure equalizing capability between the cooling water header and exhaust water header, the sequence of events required to achieve a high pressure differential between the RPV and the exhaust water header is as follows: First the RPV must be reduced in pressure. As the vessel pressure drops, the pressure in the exhaust water header will follow via a reverse flowpath through the -121 HCU valves and up to vessel through the top of the drives. Following this reduction, there must be a repressurization of the RPV facilitated without any control rods being withdrawn. As the pressure increases, a positive pressure differential results across the exhaust water solenoid valve in the HCUs and causes the solenoid valves to seal tightly against their valve seats. Since the only communication between the exhaust water header and the RPV would be through these solenoid valves, the pressure in the exhaust water header remains at the previous low pressure value of the RPV.

Due to the particular scenario of events which is required to achieve a high pressure differential between the RPV and the exhaust water header, the potential frequency of occurrence is very small. Vessel pressure decreases of the magnitude to be of interest in this discussion will result in or be initiated by the full insertion of all control rods (i.e., a scram or a controlled shutdown). From this shutdown condition the vessel pressure must then be increased without moving control rods. Such vessel pressurization may be effected directly through the use of high pressure core coolant supply systems or indirectly through the use of the recirculation pumps to heat the vessel up and consequently increase the vessel pressure. Heating the vessel using the recirculation pumps is not a normal practice. The high pressure core coolant systems are used to perform the operational hydro tests that are required following a service outage in which the vessel head has been removed.

AS:cas/92G9

c. The maximum initial control rod withdrawal speed and its potential impact on reactor safety if the equalizing valves are not installed or should become inoperative.

RESPONSE

If the pressure equalizing valves are not installed, the existing orificed check valves (i.e. valve -100) between the exhaust water header and the main header of the CRD system will continue to perform the pressure equalizing function and keep the exhaust water header pressure near the pressure of the vessel.

In the unlikely case that the pressure equalizing valves were to both fail, the maximum initial velocity of the drive would be approximately 12.2 in/sec. However, since the fluid in the exhaust water header is essentially incompressible, this rod velocity would quickly decay to normal rod speeds as the exhaust water header is pressurized. The control rod would withdraw a maximum of six inches before the exhaust water header was repressurized and normal withdrawal speeds reestablished.

This short duration of rapid control rod withdrawal will not impact reactor safety. It will only occur on the first rod selected at a time when the reactor core is subcritical.

d. The method available to verify proper operation of the equalizer valves and the recommended frequency of these verifications.

RESPONSE

The pressure of the exhaust water header is monitored and displayed locally at the CRD System flow/pressure control station. A good practice would be to verify the exhaust water header pressure prior to rod withdrawal whenever the plant has gone through the sequence of events which may lead to a high pressure differential between the RPV and exhaust water header.

Furthermore, note that there exists a simple, positive means of assuring that the exhaust water header is pressurized. By selecting a rod and giving it a continuous insert signal the insert exhaust water directional control valve for that drive will be opened and maintained open as long as the insert signal is applied. The flow to the exhaust water header during this exercise will assure that it is properly pressurized. Since the scenario of events which may lead to a high differential pressure between the RPV and the exhaust water header results in the control rods being inserted, this exercise of applying a continuous insert signal should not impact plant availability.

3. Paragraph 3.2.3 of the May 22, 1979 submittal, dealing with corrosion products, acknowledges that it is possible for corrosion products to settle in the annular clearance between the check ball and its cage. Further, if corrosion products accumulate in this space, there is no available way to detect its presence or verify that the control rods will scram other than scramming when the reactor is at an appreciable pressure.

Provide a discussion, plus supporting theoretical and experimental information, to support the statement: "There is sufficient clearance around the check ball and its cage so that corrosion products would offer little resistance relative to the large upward force exerted on the ball following a scram at elevated reactor vessel pressure." The discussion should include all critical parameters, such as the rate of crud deposition, the possibility and frequency of crud bursts, the range and distribution in sizes of crud particles, the clearance between the ball and its cage, as well as the tendency for the crud to aggregate into a somewhat solid mass with time. Further, the discussion should deal with the geometry of the surfaces which would have a tendency to further compress the aggregated crud as the ball moves up and thereby wedging the ball.

RESPONSE

The only flow to the cooling water header which passes through carbon steel piping is that flow which passes through the CRD system flow stabilizer loop. Typically, the carbon steel pipe in the stabilizer loop is composed of a two foot length of 3/4" pipe, a ten foot length of 1" pipe and about two feet of 2½" pipe. The flow through the stabilizer loop is set at 6 gpm. The minimum velocity of the flow through the stabilizer loop is therefore in the 2½" line and is about 0.54 ft/sec. The temperature of the CRD System flow is controlled within 50 - 150°F. Typical CRD cooling water conductivity is 1.0μ mho/cm measured at 25°C and the typical dissolved oxygen concentration is 5 ppm. Referring to

AS:cas/92G12

laboratory test data on carbon steel corrosion, the carbon steel corrosion particle release rate based on the aforementioned conditions, may be conservatively approximated at 200 mg/dm^2 mo. The following Table 1 provides the carbon steel corrosion test data used in this approximation. For the approximate 14 ft. of carbon steel piping in the stabilizing loop, this corrosion particle rate converts to a total release of 7.5 grams per month.

TABLE 1

CARBON STEEL CORROSION PARTICLE RELEASE RATE

Material: ASTM A179-64
Temperature: 52°C
pH: 7
O_2 Concentration: 5 ppm
Test Duration: >5000 hrs

Conductivity (μmho/cm)	Flow Rate (ft/sec)	Release Rate (mg/dm^2-mo.)
0.1	0	278
	1	12
	3	3
	6	2
3.6	0	515
	1	46
	3	52
	6	12

Reference: Brush & Pearl, Corrosion, 28, 129-135, 1972.

AS:cas/92G13

RESPONSE (continued)

Plant operating experience has been that the carbon steel corrosion particle size is very small and is much like a fine silt. This observation is supported by the following text which predicts particle size from <0.2μ to 3μ in size.

W. E. Berry, Corrosion in Nuclear Applications, John Wiley & Son, N.Y. (1971)

The potential increase in the amount of carbon steel corrosion products delivered to the drives as a result of the carbon steel in the stabilizing loop is considered negligible. The corrosion particle release rate would increase the metal particle concentration in the CRD flow by <<0.1 ppb. The metal particle content of the CRD System flow is generally about 15 ppb. Therefore, the expected contribution of carbon steel corrosion particles from the carbon steel piping of the flow stabilizer loop could increase the metal particle concentration in the cooling water flow by about 1%.

The design of the integral ball check valve is such that a minimum difference of 18 mils exists between inside diameter of the valve cage and the diameter of the ball. The inside diameter of the cage is specified as 0.900 ± 0.005 inch and the ball diameter is specified as 0.875 ± 0.002 inch.

The carbon steel corrosion particles released from the flow stabilizer loop piping are not expected to be characteristically different from those corrosion particles which are already being transported to the drives. There are no known conditions for which it is possible for the corrosion particles to form a somewhat solid mass and jam the check ball and plant operating experience does not evidence its occurrence. The crud found in the drives is better characterized as soft and silty. The potential addition of corrosion particles from the stabilizer loop is relatively so small that no change in this characteristic of the crud is expected.

AS:cas/92G14

4. It has been indicated that the potential for control rods drifting will be increased as a result of implementing the recommended CRD system modifications. A recent operating plant monthly operating report indicated the occurrence of such an event. To overcome the drifting problem, they increased the drive pressure to 400 psid.

RESPONSE

The introductory statement of question #4 is incorrect. It has never been indicated that the potential for control rods to drift would be increased by the removal of the return line. There is a potential that rod settle margin may be reduced when the return line is eliminated. A reduction in settle margin may pose an operational inconvenience but does not constitute an unreviewed safety concern.

Furthermore, the stated occurrence of a rod drift has nothing to do with the elimination of the return line. For a discussion of the conditions which cause rod drifts, refer to Table 4-7 of Technical Guide No. IV.E.1, "Operation of General Electric Control Rod Drives", USAEC, 1971.

The potential for decrease in rod settle margin results from the potential increase in back pressure offered by the exhaust water header when the return line is eliminated. The rod settle function is facilitated by the venting of the underside of the drive piston to the exhaust water header and allowing the weight of the control rod/drive piston assembly to push the drive down to the proper notch location. Since the energy available for rod settle (i.e., the weight of the control rod/drive piston assembly), is a fixed value, an increase in exhaust water header pressure will directly affect the performance of the rod settle function.

With the return line operational, the pressure at the junction of the exhaust water header and the return line is essentially defined as reactor pressure plus the small line losses of the return line flow (defned as a maximum of 5 psid). With the return line isolated (or

RESPONSE (continued)

performs the rod settle function of opening the withdraw exhaust water directional control valve. If at the end of the 5 second rod settle sequence the Rod Position Indication System (RPIS) does not detect the rod at a notch location, the rod drift alarm is annunciated. Additionally, if an unselected rod moves out of a notch position, the drift switches between the notches will sense the rod movement and annunciate the rod drift alarms. Thus, the RMCS performs both the functions of assuring that a rod is in a proper position and of detecting unselected rod movement if it occurs.

AS:cas/92G17

Describe and discuss the following:

a. The acceptability of increasing the drive pressure differential as described above.

RESPONSE

The Standard Technical Specifications for GE BWRs requires that drives which cannot be moved with normal drive water pressure be declared inoperative and continued plant operation adjusted accordingly.

AS:cas/92018

b. The potential adverse consequences that may follow from this procedure.

RESPONSE

Drives which cannot be moved with normal drivewater pressure undoubtedly have a problem. Increasing the drivewater pressure to "force" rod movement may well make the problem worse (e.g., something jammed in the drive). Increasing the drivewater pressure in itself, however, does not present a problem since the design basis of the drive is the high pressure scram.

AS:cas/92G19

c. The anticipated number of drifting control rods that will occur throughout the life of the plant as a result of implementing the final recommended CRD system modifications. The discussion should include the estimated replacement intervals of the fuel channel boxes since the frictional forces between the channel boxes and control rods is a significant parameter in the tendency for the rods to drift.

RESPONSE

Again, rod drifting will not be affected by the elimination of the return lines.

See response to 4d below for discussion of fuel channel friction.

AS:cas/92020

d. What provisions are made to verify that the control rod - channel box friction is not excessive and at what frequency should such checks be made?

RESPONSE

The standard Tech Specs require that at least once per seven (7) days all operable withdrawn control rods be notch tested. The purpose of these tests is to confirm proper drive operation. Thus, if the friction force resisting the movement of a drive becomes excessive, it will be evident in these tests and appropriate steps will be taken as defined by the Tech Specs.

AS:cas/92G21

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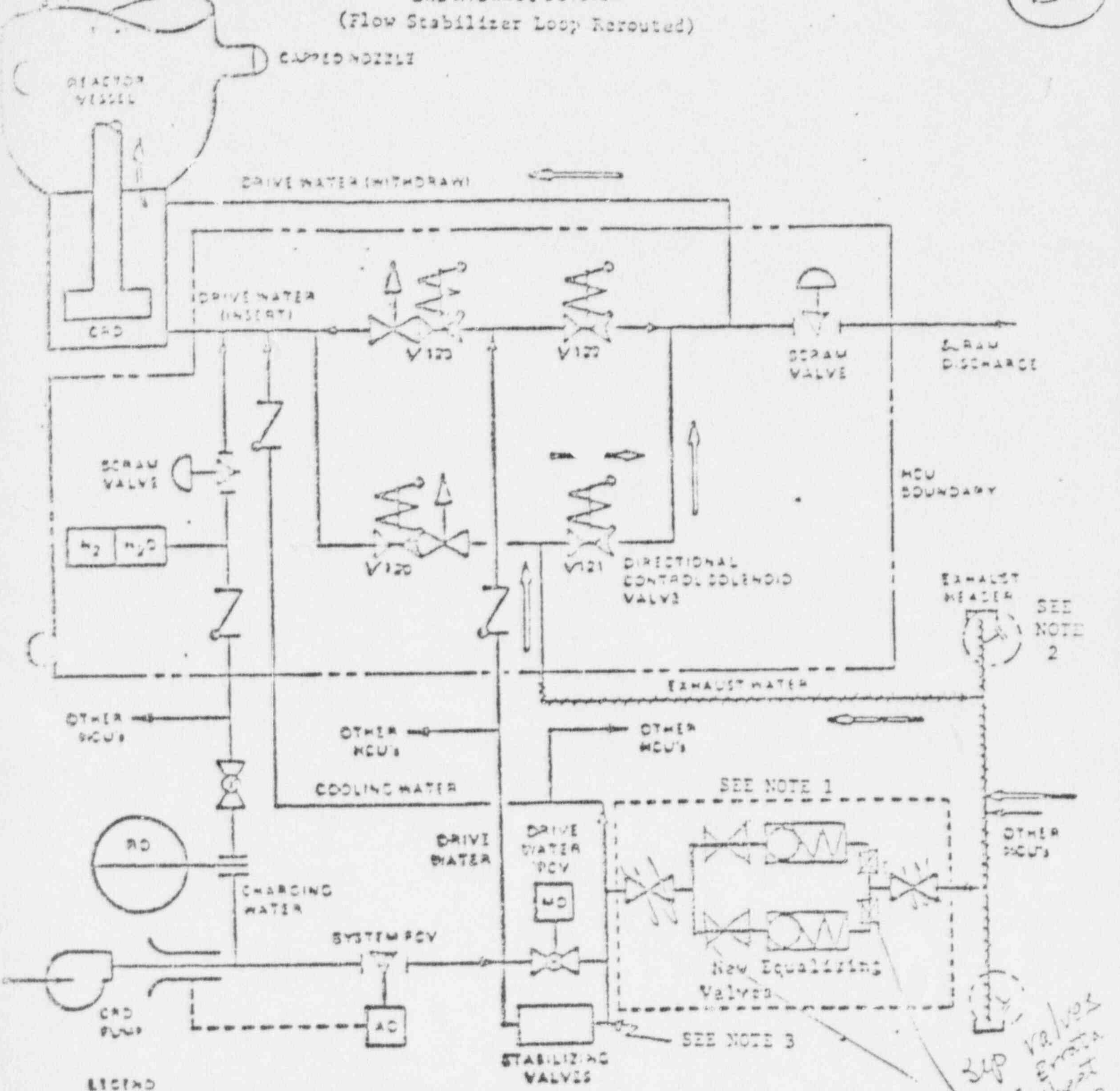
FIGURE 2

SIMPLIFIED SCHEMATIC OF MODIFIED CRD HYDRAULIC SYSTEM

(Flow Stabilizer Loop Rerouted)



LEGEND

- NORMAL FLOW PATH
- REVERSE FLOW DISPERSAL OF EXHAUST
- CARBON STEEL PIPE

Notes:

1. Pressure equalizing valves installed between cooling water header and exhaust water header.
2. Flush ports installed at high and low points of piping run if carbon steel piping in the exhaust water header is retained.
3. Carbon steel pipe in flow stabilizer loop replaced with stainless steel and rerouted directly to the cooling water header.

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U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: R. P. Snaider, Division of Operating Reactors

Gentlemen:

SUBJECT: CONTROL ROD DRIVE RETURN LINE REMOVAL

Reference: Telecon, 10/25/79

Attached are results of analyses of boil-off rates and CRD pump makeup capability use for plants not previously addressed. Also, attached is a draft procedure for optimizing CRD pump flow to the reactor.

This material is being forwarded per referenced discussion.

Very truly yours,

Glenn G. Sherwood

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Safety & Licensing Operation

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Attachments

cc: L. S. Gifford
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RELATIVE INJECTION FLOW RATES (GPM)

PLANT (No. of Drives)	ONE PUMP Present Design	ONE PUMP Proposed Design	TWO PUMP** Present Design	TWO PUMP** Proposed Design
238 BWR/6 (177)	180*	165	245	210
251 BWR/6 (193)	180*	170	245	215
218 BWR/4 (137)	175	125	305	155

* Limited by pump runout capacity

** Not a design requirement. Assumes two pump operation is viable.

PLANT (No. of Drives)	Minimum Makeup Flow to Keep Core Covered (GPM); CRD Pump is Sole Source 40 Minutes After Scram
238 BWR/6 (177)	205
251 BWR/6 (193)	215
218 BWR/4 (137)	135

(All data have been rounded off to the nearest 5 gpm)

OPTIMIZATION OF CRD SYSTEM FLOW TO THE RPV

This is a general procedure for optimizing the flow to the Reactor Pressure Vessel (RPV) from the Control Rod Drive (CRD) Hydraulic System. This procedure is applicable to CRD Hydraulic Systems with and without return lines and to systems which have a return line rerouted to the RPV via the piping of another system (e.g., RWCU, RCIC, or feedwater system).

DISCUSSION

Although providing reactor coolant makeup is not a design consideration for the CRD Hydraulic System, it is a normal, unavoidable, incidental function of the system. The CRD system flow is delivered to the RPV via the control rod drives and the CRD system return line (i.e., if the particular CRD system configuration includes an operational return line). The design bases of the CRD system flow requirements are defined to meet the needs of the system's normal functions of scramming, notching, and cooling the drives. However, if it is necessary to use the CRD system for coolant makeup, the system configuration may be adjusted to optimize the flow to the RPV. This is a recommended procedure for optimizing the CRD system flow to the RPV.

CAUTIONS AND CONSIDERATIONS

The flow capability of the CRD system varies inversely with the pressure of the RPV. In the normal system configuration, the system is adjusted so that the maximum system flow, which occurs when the RPV is depressurized, is limited to runout flow of the CRD drivewater pump. As the vessel pressure increases, a greater developed head is required from the pump and thus the CRD system

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flow decreases. The following procedure offers CRD system alterations which may be implemented to decrease the required developed head of the pumps and, thus, increase the flow to the RPV. A prime consideration when making these system adjustments is that the CRD pump runout flow of 200 gpm shall not be exceeded.

The CRD Hydraulic System is equipped with two (2) drivewater pumps. The design bases of the system are that the second pump is to be used as a backup if the primary pump were to fail or be taken out of service. If desired, both pumps _may_ be used at the same time to increase the flow to the RPV (i.e., there are no CRD system design bases which prohibit the coincident operation of both pumps). Such duo-pump operation may, however, be limited or even prohibited by other conditions such as: pump motor switch gear which does not permit two-pump operation; load limiting breakers on the pump motor power supplies which may limit the electrical power available for two-pump operation and, hence, two-pump flow capability; and/or, common pump suction piping losses which may cause one or both pumps to trip due to low net positive suction head (NPSH). Therefore, it is suggested that the electrical power supply and switching for the pumps and the pump suction piping line losses be evaluated for adequacy prior to operating the CRD system with both pumps.

RECOMMENDED PROCEDURE

Keeping in mind that the maximum pump runout flow of 200 gpm per pump should not be exceeded, the general procedure for optimizing the CRD system flow to the RPV is as follows (refer to Figure 1 for subject valve locations):

1) Scram the reactor. This will open the scram injection valves (HCU valves 126) and provide a relatively free flowpath to the drives;

2) Open all valves downstream of the main flow control valve (F002), such as the drive water pressure control valve (F003) and its manual bypass valve (F004). If the system has a return line, open all valves on the return line (e.g., valves F005, F006, and F032). If the system has pressure control valves on the cooling water header, open both the air-operated (F127) and manual-operated (i.e., bypass) valve (F128).

3) While monitoring the pump discharge flow* to assure that its maximum runout flow of 200 gpm is not exceeded, slowly inch open the main flow control valve (F002A).

4) If the flow control valve is fully opened and yet additional flow to the RPV is desired, the backup flow control valve (F002B) may also be put into service by opening valves F046B and F047B.

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*Monitoring of pump discharge flow may be accomplished by determining the developed head of the pump from the pump suction and pump discharge pressure indicators (R017 & R008, respectively) and referring to vendor supplied pump performance data for the corresponding approximate discharge flow. Alternatively, the flow transmitter (N004) on the flow element for total system flow (N003) may be respanned to accommodate higher flows. However, this flow reading would not include the pump minimum bypass flow, the flow in the pump test bypass line (if any), or the recirculation pump seal purge flow (if any). These additional flows must be considered if the reading from flow element N003 is used to determine the pump discharge flow.

5) For additional flow, depending on system design, valves between the drivewater pump and the flow control station such as the pump discharge pressure (F170) and pump test bypass line isolation valve (F018) may be opened. The backup drivewater filter may also be brought into service by opening valves F020B and F021B. (Closing the minimum flow bypass valve, i.e., F015 A&B is not recommended.)

6) For the two-pump operation, first close the pump discharge isolation valve (F014B) of the second pump (this should reduce the potential for the pumps to trip on low NPSH when the second pump is started). Start the second pump. Slowly inch open the F014B valve. With valve F014B fully open and both pumps running, the developed head across the pumps will still indicate the approximate flow through each pump (this of course assumes that both pumps performance are alike and well represented by the vendor supplied performance data).

7) Depending on system design, the pump NPSH may be improved during two-pump operation by opening the pump suction filter bypass line (i.e., valve F117 and F116).

Again, particular attention must be given to the pump discharge flow if any of these actions are taken to optimize the flow to the RPV. Consider the fact that reducing vessel pressure will lead to increasing CRD system flow when the system is in the scram mode. Therefore, if the system flow is increased to near runout flow at a given vessel pressure and the vessel pressure subsequently drops, actions (i.e., valve adjustments) will be necessary to assure that the concurrent flow increase does not result in a flow greater than the defined 200 gpm per pump.

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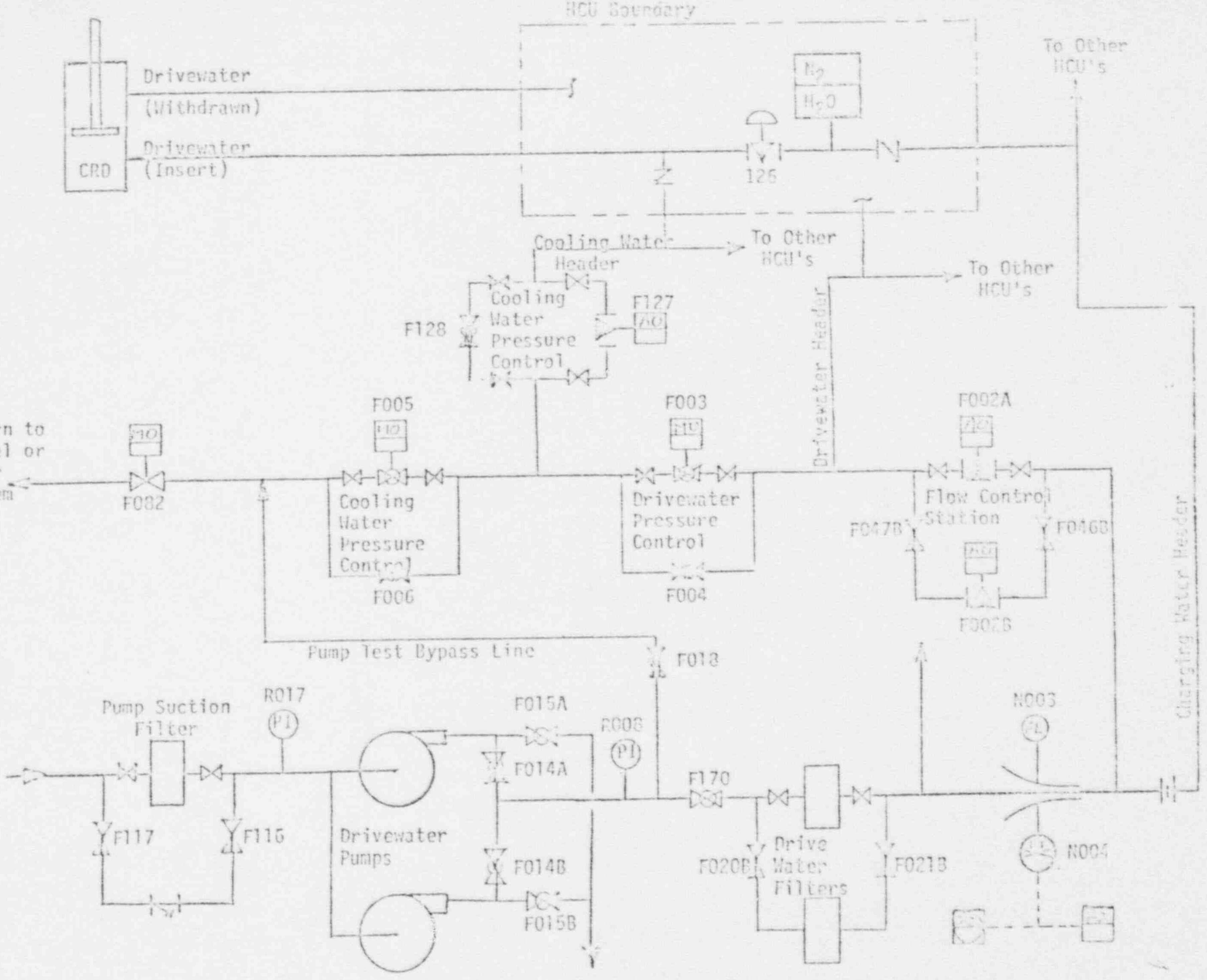

FIGURE 1. COMPOSITE REPRESENTATION OF TYPICAL CONTROL ROD DRIVE HYDRAULIC SYSTEM

ICSB-17 RPS MG SET MODIFICATION DESIGN

The Shoreham RPS MG SET design has been modified by the addition of redundant, Electrical Protection Assemblies (EPA) to preclude the possibility of unacceptable RPS MG SET performance under degraded voltage conditions. The EPS design is the same for all BWR's.