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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 18 1984

MEMORANDUM FOR: Harold R. Denton Director
Office of Nuclear Reactor Regulation

Richard C. DeYoung, Director
Office of Inspection and Enforcement

FROM: C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

SUBJECT: CASE STUDY REPORT - EDWIN I. HATCH UNIT 2
PLANT SYSTEMS INTERACTION EVENT OF AUGUST 25, 1982

The Office for Analysis and Evaluation of Operational Data has finalized its case study report for the Edwin I. Hatch Unit 2 plant systems interaction event which occurred on August 25, 1982. The final report replaces the preliminary case study report and reflects the comments provided during peer review. We have enclosed a copy of our final report for your information and appropriate action.

Based on our detailed study, we have concluded that the Hatch event was a significant operational occurrence in several important respects. First, the event involved a sustained and uncontrolled leakage of hot pressurized reactor coolant outside primary containment which lasted for several hours. Additionally, the Hatch experience provides a clear illustration of the potential for the BWR scram system to cause such an occurrence. The event was also noteworthy in that four independent random failures were required to cause the sustained loss of reactor coolant outside containment. And finally, the event was important in that the corrective measures that could have prevented most of the key equipment failures were discussed in NRC or industry operational experience feedback correspondence several years before the event occurred.

In general the report concludes that the Hatch event, including its causes and consequences, is bounded by the postulated event scenarios generically evaluated in NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping." It is also concluded that, if the staff positions and guidance which followed the generic evaluation were implemented on a plant-by-plant basis, adequate mitigation capability would be assured for both the Hatch event and the more limiting postulated accident sequences. At the same time, we are aware that the staff has not yet finalized its positions for all of the outstanding issues associated with NUREG-0803, including the need for

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environmentally qualifying the needed mitigation equipment. We would anticipate, therefore, that the Hatch event will be carefully considered in formulating the final staff positions and/or proposed requirements for resolving the outstanding issues.

Regardless of the staff's final determinations, the enclosed case study report contains several specific recommendations and suggestions which relate to the individual equipment failures and problems which occurred during the Hatch event. We believe that these recommendations and suggestions should be carefully evaluated for follow-up actions by NRR, IE or an appropriate industry group as outlined in the report. We understand that some of the recommendations and suggestions are already being evaluated by groups either inside or outside the NRC as a result of their earlier appearance in the preliminary case study report. Some of the recommended or suggestion actions may in fact be nearing completion. In any case, we would urge NRR and IE to continue to pursue completion of the appropriate follow-up actions as discussed in our report.

AEOD would be pleased to provide any additional information or clarification which you or your staff may require. Please contact either Karl V. Seyfrit (492-4440) or Stuart D. Rubin (492-4436) if you or any member of your staff has any questions concerning the final report.

C. J. Heltemes, Jr.

C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
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Enclosure:
As stated

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CASE STUDY REPORT

for the

EDWIN I. HATCH UNIT NO. 2

PLANT SYSTEMS INTERACTION EVENT

on

AUGUST 25, 1982

by the

OFFICE FOR ANALYSIS AND EVALUATION

OF OPERATIONAL DATA

MAY 1984

Prepared by: Stuart D. Rubin

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Note: This report documents results of a study completed to date by the Office for Analysis and Evaluation of Operational Data with regard to a particular operational situation. The findings and recommendations do not necessarily represent the position or requirements of the responsible program office nor the Nuclear Regulatory Commission.

ABSTRACT

A study was performed for a plant transient which occurred at the Hatch Unit 2 reactor facility on August 25, 1982. The complex series of systems interactions which occurred during post-scram recovery operations resulted in a sustained and uncontrolled loss of hot pressurized reactor coolant outside primary containment. The study concludes that the positions and guidance developed from a recently concluded generic review of a similar postulated event are adequate to address the safety concerns associated with the actual Hatch event and its consequences. However, the event could have been prevented had adequate corrective actions been taken in response to the lessons learned from prior operating experience. Followup corrective measures are suggested to address the specific areas needing attention.

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EXECUTIVE SUMMARY

A study was performed to evaluate a plant transient that occurred at Edwin I. Hatch Nuclear Plant Unit 2 on August 25, 1982. The event began when a main steam isolation valve failed closed initiating a reactor scram and vessel isolation. During post-scram recovery operations a sustained and uncontrolled reactor coolant system blowdown occurred outside primary containment. The coolant lost from the reactor, exited via the control rod drives and discharged into the reactor building equipment drainage system through a partially stuck open drain line isolation valve on the scram exhaust volume. The scram exhaust volume was maintained in a hot pressurized condition by the reactor for several hours after the reactor scrammed because of a prolonged high drywell pressure trip condition which could not be cleared or reset by the operators. High temperature reactor coolant was released to the open areas of the reactor building through an open equipment drain hub located in the reactor core isolation cooling (RCIC) diagonal room in the basement of the building. The local environment was sufficiently harsh to shut down the operating RCIC system and set off the fire suppression system. The local ambient temperature some distance away from the RCIC room exceeded the qualification temperatures for the vital electrical power supply equipment located there. Eventually the emergency bypassing of signals by operating personnel outside the control room was required to reactivate the cooling equipment needed to depressurize the drywell for scram reset and termination of the blowdown. At no time during the event was there a danger of inadequate core cooling or inadequate core cooling capability, however. The event would appear to be significant in that it may mark the first time that a domestic commercial boiling water reactor nuclear power plant has experienced a prolonged uncontrolled blowdown of the reactor coolant system outside primary containment during hot pressurized conditions.

The assessment provided in the study concludes that the Hatch event can be viewed as a "precursor" for a similar but more limiting postulated accident sequence that has recently been comprehensively reviewed on a generic basis by the NRC staff. The study further concludes that if the staff positions and guidance which resulted from the earlier generic review are implemented on a plant-specific basis, adequate preventive and mitigation measures will have been provided for both the Hatch event and the more limiting postulated accident scenario.

Nevertheless, the underlying causes for a number of the specific equipment failures and problems which occurred during the Hatch event were found to be significant in that they were addressed in official NRC correspondence transmitted to the Hatch licensee (Georgia Power Company) and other boiling water reactor (BWR) licensees years before the event occurred. The earlier communications, which addressed the main steam isolation valve, scram discharge volume isolation valve, and equipment drain hub covers, contained substantial information relating to the causes and needed corrective actions for the problems associated with these components, and were prompted by earlier reviews of prior similar or related operational experiences at the Hatch plants and/or other BWR facilities. The study thus concludes that any one of these equipment problems could have been prevented and the significant plant response consequences avoided had adequate corrective actions been implemented in response to these communications. To correct this situation followup corrective measures have been suggested which address several of the specific areas that appear to be in need of attention.

Finally, the Hatch event underscores the potential for the reactor building equipment and floor drain systems to channel adverse environments to distant areas of the reactor building. The study recommends that a review be performed to evaluate the potential for the floor drain system to channel harsh environments (associated with high energy line breaks outside containment) to vital areas of the reactor building which are otherwise protected against such harmful conditions.

1. INTRODUCTION

On August 25, 1982, the Edwin I. Hatch Nuclear Plant Unit 2 experienced a system transient which resulted in a reactor trip and reactor vessel isolation from rated power conditions. During the post-scram recovery phase of the event a series of equipment failures, problems, and systems interactions occurred that resulted in a sustained uncontrolled and unisolable blowdown of reactor coolant outside primary containment. Coolant lost from the reactor exited via the control rod drives through a partially stuck open isolation valve in the scram discharge volume (SDV) piping system. Emergency bypassing of protection signals by plant personnel, at locations outside the control room, were required to terminate the discharge of primary coolant directly into the open areas of the reactor building. The adverse environment in the reactor building which resulted from the blowdown shut down the reactor core isolation cooling system, which was providing coolant makeup to the vessel at the time. The event also resulted in a significant increase in the ambient temperature in parts of the reactor building some distance from the point of release, and actuation of the reactor building fire suppression system.

This report provides the results of an investigation of the event by the Office for Analysis and Evaluation of Operational Data, USNRC. Section 2 provides a detailed description of the sequence of events involved in the initial plant transient and the post-scram recovery. Included are the time history of the major events, significant operator actions, and important plant personnel activities. Section 3 provides a brief description of some of the systems that played key roles in determining the event consequences. Section 4 discusses the principal equipment failures and problems which occurred during the event. The cause for the failure or problem and the corrective actions taken or planned by the licensee are also provided. Section 5 contains an analysis and evaluation of the event from an overall integrated event sequence viewpoint, with regard to the significant individual equipment failures and problems that occurred. The overall integrated event is assessed relative to a similar but postulated accident scenario recently reviewed by the NRC staff on a generic basis A discussion is also provided for the key systems interactions that occurred and resulted in the significant plant response of a prolonged blowdown of reactor coolant outside primary containment. The individual equipment failures and problems which occurred during the event are also discussed in relation to both similar prior experiences at other facilities and relevant prior NRC communications with BWR licensees. Section 5 also provides a description of the potential for the reactor building floor drain system to channel harsh environments to separated vital areas during a postulated high energy line break outside primary containment. Section 6 presents the principal findings and conclusions that resulted from the investigation, including the analysis and evaluation of the information collected. Section 7 provides recommendations for followup actions which could be taken to address the areas of concern discussed in Sections 5 and 6.

2. EVENT DESCRIPTION

At approximately 4:17 a.m. EST on August 25, 1982, Edwin I. Hatch Nuclear Plant Unit 2 sustained a reactor scram and Group 1 isolation from full power conditions (Ref. 1). The event was initiated when the inboard isolation valve on the "C" main steam line closed unexpectedly (Refs. 2, 3, 4 and 5). The resulting steam flow shutoff in the "C" steam line caused a marked increase in reactor pressure which led to a high neutron flux scram due to core void collapse. At the same time, steam flow redistribution to the three steam lines which remained open caused a Group 1 isolation (automatic closure of all main steam line isolation valves) from a high steam flow condition in these lines. With the reactor scrammed and isolated, vessel pressure increased rapidly towards the opening pressure of the safety/relief valves. As pressure increased to about 1090 psig the "D" safety/relief valve (SRV) lifted automatically to relieve steam. As is the normal procedure during such transients, the control room operators went to manually open the "H" SRV to increase the vessel blowdown rate which would reduce pressure further. The "H" SRV did not lift when its control switch was turned to the open position, however. When the "H" SRV failed to open manually, the operators went to actuate the "A" SRV to assist in pressure control. The "A" SRV successfully lifted at this time and reactor pressure was brought back down to approximately 900 psig.

The reactor scram and vessel isolation also resulted in an expected rapid shrinkage of vessel water level. Level dropped to the low-low level setpoint, initiating both the high pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system. However, the combined effects of injection flow coast-down from the turbine driven reactor feed pumps and SRV operation quickly raised water level back up to the high level trip setpoints for HPCI and RCIC. Accordingly, even though both systems auto-started, no injection into the vessel actually occurred prior to their tripping off-line.

With vessel water level restored and pressure stabilized, the control room operators prepared to reopen the closed main steam isolation valves (MSIVs). The operators first reset the Group 1 isolation signal which had already cleared. Isolation reset allowed pressure equalization around the closed MSIVs via the main steam line drains which had also isolated during the event. Pressure equalization across the MSIVs was begun at 4:20 a.m. Once all of the initial reactor trip conditions cleared, the operators reset the scram signal which allowed the scram discharge volume to begin to drain and depressurize. By this time the RCIC system was manually restarted for level control of the isolated vessel. At 4:29, inventory loss through the open main steam line drains resulted in a low reactor water level alarm condition even though RCIC was operating. When this occurred HPCI was manually restarted to restore water level. By 4:32, reactor water level and pressure were again stabilized at 32 inches (normal operating level is 33 inches) and 990 psig. Scram recovery operations continued in this manner while pressure equalization around the closed MSIVs continued.

In the drywell, pressure rose gradually from slightly less than 0.5 psig immediately after the reactor scrammed to about 1.0 psig 30 minutes later. During this period, the control room operators were most concerned with maintaining both reactor pressure and level. The operators manually opened the "A" SRV a second time at 4:49 a.m. to reduce reactor pressure and to facilitate pressure equalization around the closed main steam isolation valves. At 4:50, with pressure equalized, the MSIVs were successfully reopened by the operators.

Immediately after the "A" SRV was opened for the second time, the operators observed drywell pressure increasing rapidly. Drywell pressure rose above the 2.0 psig high pressure scram setpoint, and reached 2.7 psig at about 4:51 a.m. High pressure in the drywell initiated a second reactor scram (the control rods had already fully inserted following the first scram), and several primary containment isolations which could not be reset by the operators. The high drywell pressure signal also caused the drywell chiller units and control rod drive pumps to trip. This occurred by design from load shedding logic associated with the emergency buses which supply power to these systems. Loss of the chillers interrupted normal drywell cooling at this time. Attempts to manually restart the chiller units were unsuccessful due to the loss of electrical power caused by the load shedding logic. Pressure in the drywell continued to rise and reached approximately 4.0 psig at about 4:57. The loss of the control rod drive (CRD) pumps also resulted in a loss of cooling flow to the CRDs. As a result, CRD seal temperatures started to increase beyond the normal 160°F to 200°F range. This was indicated by the control rod drive high temperature alarm that sounded in the control room about this time.

At 5:10 a.m., the RCIC system isolated on a high turbine exhaust diaphragm pressure signal while it was injecting into the vessel. Several attempts by the control room operators to restart the RCIC system proved unsuccessful. At 5:15, the 2A reactor feed pump was restarted to provide reactor coolant makeup to the vessel.

About this time (just before 5:25 a.m.) a high temperature alarm was received from the RCIC room located in the northwest (NW) corner of the reactor building basement. Indications also were received that the RCIC room deluge system had actuated. Additionally, health physics personnel working in the reactor building reported "smoke" coming out of the RCIC room. However, operating personnel soon determined that the "smoke" was actually steam rising up the RCIC corner room stairwell. Once operating personnel verified that no fire actually existed, the deluge system was secured. Plant personnel also observed steam vapor rising up the stairwell from the southwest (SW) corner where the reactor building equipment drain sump is located. This steam and hot air, along with the steam and hot air rising from the RCIC corner roc caused ambient temperature on the 130' elevation (i.e. the floor immediately above the 87' basement elevation) to increase. Air temperature around the CRD hydraulic control units located on the 130' elevation in the reactor building rose to about 130°F. During this time, CRD temperature instrumentation indicated that drive temperatures had increased to over 500°F due to the earlier loss of cooling water flow from the CRD pumps. This was well beyond the normal operating temperature range of 160°F to 200°F.

Operating personnel observed that fluid temperature and level in the reactor building equipment drain sump, located in the SW corner room, was rising well beyond normal operating values. The equipment drain sump pumps initially attempted to cool the sump fluid by operating in the recirculation mode. However, the rate of influx of fluid into the sump necessitated pumping the rising hot fluid out of the sump to the liquid waste collection tanks located in the radwaste building. During this time, considerable amounts of hot water also were being pumped out of the adjacent reactor building floor drain sump located in the same SW corner room.

Based on the overall indications in the reactor building, operating personnel concluded that hot scram exhaust water from the still pressurized reactor, was discharging at high pressure into the reactor building equipment drainage system. To terminate the discharge of high temperature fluid into the reactor building, the control room operators realized that it would be necessary to reset the scram. However, the high drywell pressure scram condition which existed could not be reset until actual drywell pressure could be reduced below 2.0 psig. The operators knew that rapidly decreasing drywell pressure by normal means was precluded, since the chillers had been lost earlier in the event by the load shedding logic initiated by the same high drywell pressure condition. High drywell pressure had also isolated the main and bypass exhaust lines of the primary containment ventilation system. Venting with this system is an alternate means that can be used to reduce drywell pressure. Accordingly, with adequate inventory makeup provided for reactor level control, the operators turned their priority attention to rapidly depressurizing the drywell below the 2.0 psig scram setpoint.

As a first step, the operators activated the high drywell pressure override switches for the isolation valves installed in the small diameter bypass lines associated with the drywell main ventilation lines. Following this action, the bypass lines were opened. This established a limited bleed-off path from the drywell. The control room operators immediately observed, however, that drywell pressure was remaining high and was appearing to drop only very slowly. Realizing that it would take many hours (if not days) using this depressurization scheme to reduce pressure below the 2.0 psig scram, load shedding, and containment isolation setpoints, the control room operators considered what alternative actions might be taken to quickly reset the scram.

The operators decided to dispatch a technician into the reactor building to begin to take steps to bypass the high drywell pressure signal to the trip circuit of the circuit breakers for the drywell chiller units. Once this could be achieved it would be possible to restart the drywell chillers. Operation of the chillers would allow drywell pressure to be brought back down below 2.0 psig. This would enable the operators to reset the high drywell pressure scram. Once the scram was reset, the discharge of hot scram exhaust water into the reactor building would be terminated. The operators recognized, however, that it might take several hours before this could all be accomplished.

In the meanwhile, operating personnel were able to reestablish power to the tripped CRD pumps by bypassing the high drywell pressure load shedding logic associated with the pump motor electrical supply. With power to the pumps restored, operating personnel were in a position to restart the CRD pumps to reestablish cooling water flow to the CRD seals. Plant personnel elected not to start the CRD pumps at that time, however. The pumps were not restarted immediately by the operating staff because of their concern for possibly causing seal damage if relatively cool water from the condensate storage tank were introduced into the very hot CRD seals.

While operating personnel waited for the electrical technician to bypass the trip condition to the drywell chillers, steam and hot air from the NW corner and SW corner rooms of the reactor building basement continued to flow up the stairwells into higher elevations. During the period when reactor water was being lost through the CRDs and the scram discharge volume headers, the control room operators continued to maintain reactor pressure and level with the main condenser and feedwater systems. At 6:30 a.m., the high drywell pressure signal to the trip circuit of the circuit breaker for the drywell chiller units was successfully bypassed. The chillers were restarted at this time. As soon as the chillers were placed back into operation, control room personnel observed actual drywell pressure starting to decrease.

About 7:30 a.m., one of the CRD pumps was restarted, after operating personnel had consulted with General Electric, the nuclear steam supply system vendor about the potential for causing CRD seal damage. As indicated by local temperature recorders located in the reactor building, this action brought CRD seal temperatures from over 500°F back down to normal operating values. No indication of CRD seal degradation was observed when seal temperatures returned to normal operating conditions.

Finally, at approximately 7:40 a.m., or about 2 hours and 45 minutes aft. the high drywell pressure condition and blowdown into the reactor building initially occurred, the drywell chillers successfully reduced drywell pressure back down below the high drywell pressure reactor trip setpoint. As soon as the high drywell pressure condition cleared, control room personnel reset the high drywell pressure scram. This action reclosed the open scram outlet valves, stopping the flow of hot (reactor) water and steam into the reactor building equipment drain system and basement corner rooms.

At this time operating personnel proceeded to bring the reactor to a cold depressurized shutdown condition. Several hours later, plant personnel went down into the RCIC room to assess radiation and contar iation levels and to prepare to begin their investigation of the event, it. causes, consequences and needed corrective actions.

3. SYSTEMS DESCRIPTIONS

This section provides a brief description of the design and operation of some of the plant systems which were involved in the Hatch Unit 2 event. The descriptions reflect the equipment designs at the Hatch plant as they existed on the date of the event. A more complete discussion of these and other systems may be found in the Edwin I. Hatch Unit No. 2 Final Safety Analysis Report (Ref. 6).

3.1 Scram Discharge Volume System

The purpose of the scram discharge volume (SDV) system is to receive, contain and limit the water exhausted from the reactor via the control rod drives during a reactor scram. The SDV system, shown in simplified schematic form in Figure 1, consists of the SDV headers, interconnected piping, and associated valves. At Hatch Unit 2, water exhausted from individual CRDs is piped to and through the associated individual CRD hydraulic control units (HCUs). From there it is routed to one of two banks of header piping located inside the reactor building secondary containment on opposite sides of the reactor vessel. Both the hydraulic control units and the SDV system are located outside of the primary containment structure. As shown in Figure 1, each CRD is connecte l to one of the two SDV headers via a scram outlet valve mounted within its associated HCU.

At Hatch Unit 2 each of the two SDV header banks has an instrumented volume attached directly below the header piping. Rising above the high points of each bank of header piping are small diameter vent lines equipped with a single normally open vent line isolation valve which automatically closes on a reactor scram signal. Both vent lines are routed to and hard-piped into a nearby branch line of the embedded reactor building equipment drainage system piping network. At Hatch Unit 2 a drain line is also connected to the bottom of each of the two instrumented volumes. The two drain lines come togeth r to a single line equipped with a single normally open drain line isolat on alve. The SDV vent and drain valves are normally open during reactor power o ation to allow any water which might enter the SDV headers to continuously dr.. out of the system. Any water that flows through the SDV drain line is received via a hard-piped connection to a local embedded piping branch of the reactor building equipment (clean radwaste) drainage system.

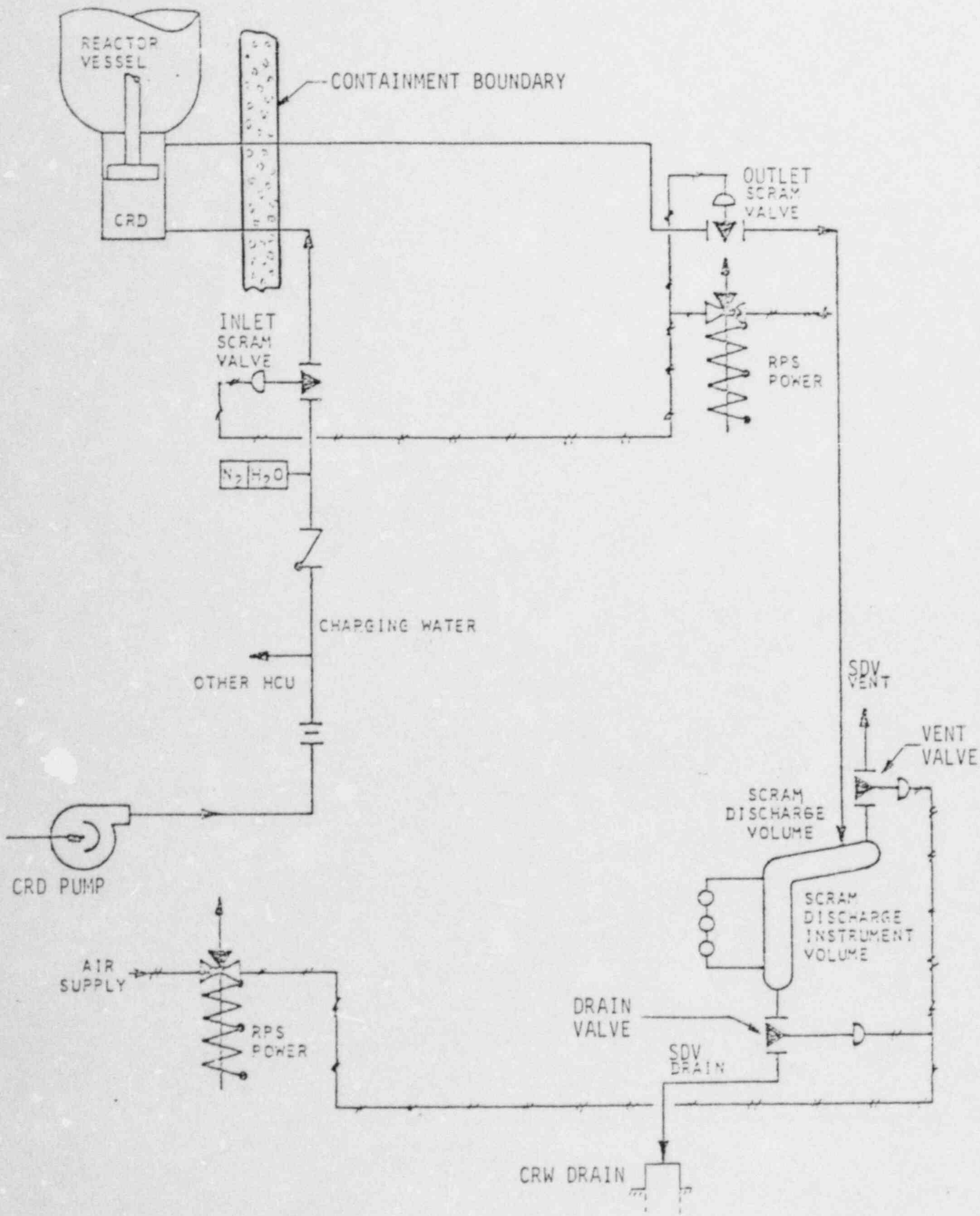


Figure 1 Control Rod Drive/Scram Discharge Volume Systems

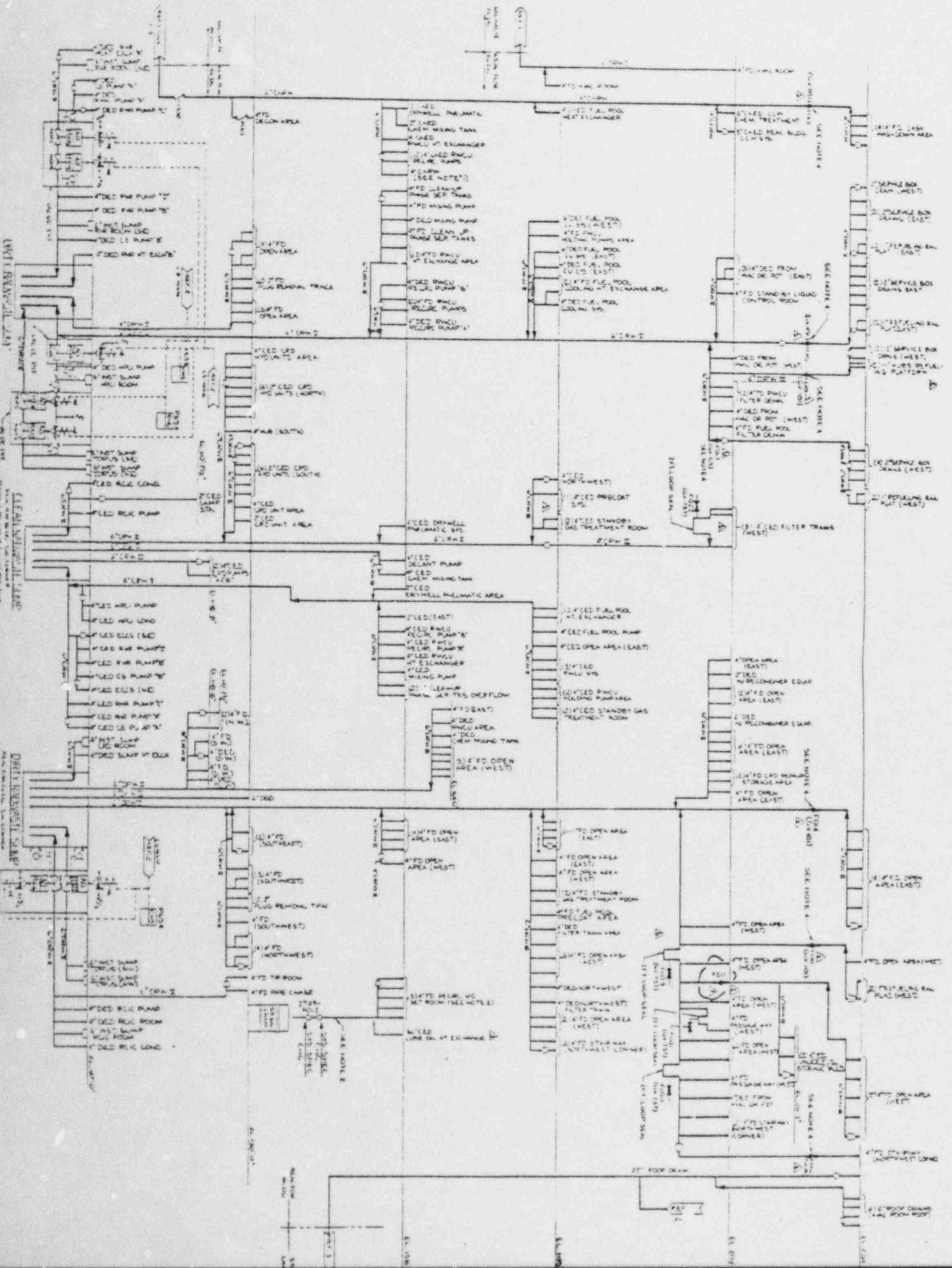
As shown in Figure 1, the scram pilot air solenoid valve(s) control the scram inlet and outlet valves of each CRD via the reactor protection system (RPS). The RPS also controls the vent solenoid valve(s) which pilot open and close the SDV system vent and drain line isolation valves.

Upon a reactor scram (initiated by an RPS trip condition), the individual scram inlet and scram outlet valves open, and the SDV drain and vent valves close. As a result, the SDV system piping fills and pressurizes as it accepts, contains, and limits the water exhausted from the reactor through the control rod drives. Even after the control rods have fully inserted, with the scram valves left open, reactor coolant continues to flow past the CRD seals, through the scram outlet valves and into the SDV system piping pressurizing it to full reactor pressure. Thus, during and immediately following a scram the SDV system becomes a reactor coolant retaining boundary outside primary containment. The integrity of the SDV system during this period is dependent upon full closure of its vent and drain line isolation valves. These valves seal the volume to prevent continued release of pressurized reactor coolant into the reactor building equipment drainage system during and immediately following a reactor scram. After a manual or automatic reactor scram, four float-type level switches located in the instrumented volume, and which interconnect with the trip channels of the RPS, will open to initiate another reactor scram signal. These switches are provided to initiate a reactor scram should water accumulation attempt to fill the instrumented and scram exhaust header volumes during normal plant operation. A handswitch is provided on the reactor control panel in the control room to bypass the trip function of the four level switches when the reactor mode switch is in either the shutdown or refueling positions. his permits the discharge volume vent and drain valves to be reopened, and the scram inlet and outlet valves to be reclosed after the RPS is reset following a reactor scram. This action enables the SDV to be drained following a reactor scram without initiating a subsequent scram due to an SDV high water level signal. However, if a scram condition is present which cannot be reset or bypassed with the mode switch in either the shutdown or refueling position, reclosure of the scram outlet valves by RPS reset with the bypass handswitch is prevented.

3.2 Reactor Building Floor Drainage System

The purpose of the reactor building floor drainage system is to collect radioactive and/or nonradioactive liquid wastes spilled or released onto the floors of the various rooms and elevations of the reactor building and to route the liquid to central collection points for removal to a suitable disposal area. Flow of liquids into the floor drainage system is received through open floor drains located in the various equipment areas of the reactor building. As shown in Figure 2, liquids received by the floor drains are collected in branch lines, emptied into main waste lines and discharge into one of two reactor building floor drain sumps. At Hatch Unit 2, one of the sumps is located in the SW corner room of the reactor building basement while the other sump is located in the southeast (SE) corner room of the basement. At the basement level, the open floor drains in the HPCI room, the northeast (NE) corner (RHR and core spray) room, the SE corner (RHR and CS) room, and the central (torus) room

Reactor Building Floor and Equipment Drainage Systems Flow Diagram



discharge into the SE corner room sump (refer to Figure 3). Similarly, the open floor drains in the NW corner (RCIC) room, the SW corner room and the torus room discharge into the other sump located in the SW corner room. The installation of the imbedded collection piping provides a uniform slope which induces the liquid waste to flow in and thereby drain the piping to the sumps. The reactor building floor drain sumps are each provided with two 50 gal/min sump pumps. The sump pumps are started or stopped on a rise and fall of the sump level. One pump serves as a backup to the other. An abnormally high level in the sump is also alarmed in the control room by a level switch.

In addition to the two main floor drain sumps located in the SE a d SW corner rooms, each of the equipment rooms in the basement is equipped with a smaller intermediate floor drain sump. The smaller local sumps are instrumented with float-type level switches that can sense an increase in floor liquid level in any of these rooms. The floor drains in any of these basement rooms may be isolated from the associated main floor drain sump by means of isolation valves located in the branch lines. At the Hatch plants, the isolati.n valves are normally open to allow continuous drainage into the main floor drain sumps. The isolation valves will close automatically if liquid level in the local sump rises sufficiently to activate the level switch. This action prevents a potential local flooding condition from spreading to a nearby equipment room via the floor drainage system. That is, valve closure prevents common mode flooding of separated and redundant vital equipment. The isolation valves may also be closed manually from the control room. During normal operation, the branch lines and local sumps are empty and dry.

3.3 Reactor Building Equipment Drainage System

The purpose of the reactor building equipment drainage system is to collect radioactive and/or nonradioactive liquid wastes, originating as equipment drain leak-offs in the reactor building, and to route the liquid to a central collection point for removal to a suitable disposal area. As shown in Figure 2, equipment wastes are collected in a closed piping network consisting of branch lines which empty into main waste lines that converge and discharge to a single equipment drain sump. At Hatch Unit 2, the equipment drain sump is located in the SW corner of the reactor building basement. The embedded collection piping is layed with a uniform slope which induces the collected liquid waste to continually drain to the sump. The reactor building equipment drainage system does not incorporate automatic or manual valves to isolate branch portions of the system from the rest of the piping network. Isolating sections of the system is unnecessary since the piping system is effectively closed (sealed) with respect to the surrounding open areas of the reactor building. Thus, backflow flooding or liquid release to one or more equipment rooms would normally not be possible.

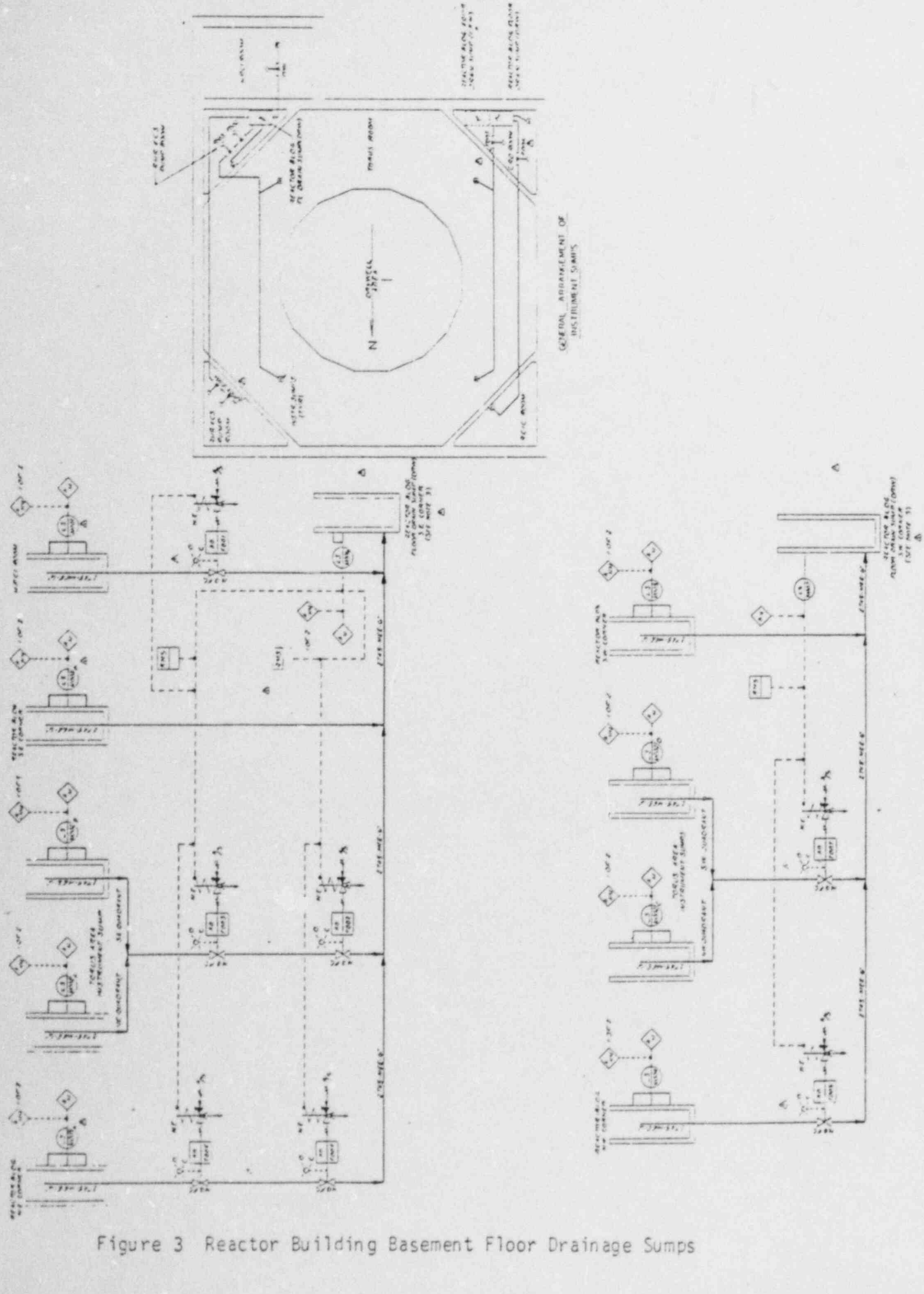


Figure 3 Reactor Building Basement Floor Drainage Sumps

At the Hatch facility, the reactor building equipment drainage system incorporates drain hubs with removable threaded steel covers that are located at floor level in the various equipment rooms of the reactor building basement. The covers may occasionally be removed to open the drainage system for temporary access during the performance of equipment calibrations, testing or maintenance activities.

The equipment drain sump is provided with two 50 gal/min sump pumps which operate in a manner similar to the sump pumps provided for the floor drain sumps discussed in Section 3.2. Water collected by the sump may be passed through a cooler when necessary. A high level in the sump is alarmed in the control room by a level switch. At Hatch Unit 2, the reactor building equipment drain sump in the SW corner room is located immediately adjacent to the reactor building floor drain sump. The two sumps are cross-connected by a penetration in the adjacent side of the two sumps. Thus, in the event of an abnormally high level in the equipment drain sump, liquid will automatically transfer to the floor drain sump by the cross-connected overflow line.

4. CAUSES AND CORRECTIVE ACTIONS

This section discusses the causes that were found for the principal equipment failures and the related systems problems which occurred during the event at Hatch Unit 2. The short term corrective actions which were taken in order to ready the plant for its return to power are also included.

4.1 Main Steam Isolation Valve Failure

To determine the cause of the failure of the "C" steam line inboard MSIV, the failed valve was removed, disassembled and inspected by both the licensee, Georgia Power Company, and the valve manufacturer, Rockwell International (Ref. 7). An examination of the internal parts removed from the valve showed that separation of the valve disk from the stem had caused the valve to go closed unexpectedly while the plant was operating at full power. Disk separation was traced to an improper stem-to-disk poppet thread engagement which allowed the poppet and disk to slip off. The entire disk and stem assembly were replaced in both the inboard and the outboard isolation valves on the "C" steam line.

4.2 Safety/Relief Valve Failure

The "H" SRV failure was investigated by the licensee following the event (Ref. 8). The failure of the valve to open, when manually actuated by the control room operator, was attributed to a component failure within the manual handswitch located on the control room panel board. The malfunction of the handswitch was traced to worn parts within the switch mechanism. Following this determination the faulty handswitch was replaced.

4.3 High Drywell Pressure and Safety/Relief Valve Tailpipe Vacuum Breaker

The cause for the pronounced and unexpected increase in drywell pressure beyond the high drywell pressure scram setpoint was also investigated by the licensee following the event (Refs. 3, 4, 5, and 8). The rate and magnitude of the drywell pressure increase that occurred during the event should not normally be expected for a reactor scram involving a Group 1 isolation, even if multiple SRV actuations occur. Added heat inputs to the primary containment result from the SRV steam blowdowns, the HPCI and RCIC turbines exhausting to the suppression pool, and the elevated SRV tailpipe temperatures. These heat sources should not increase containment pressure to the high drywell pressure setpoint, however, and should be adequately accommodated by the heat removal capacity of the drywell chiller unit. For this event these chiller units operated up to the time when the high drywell pressure condition occurred.

A review of primary containment pressure recorder data following the event showed that drywell pressure rose relatively gradually over the first 30 minutes following the initial reactor scram (Refs. 4 and 5). The data show that just prior to the first scram, drywell pressure was approximately 0.4 to 0.5 psig. Furthermore, drywell pressure had risen gradually to only about 0.9 to 1.0 psig 30 minutes later. This was still well below the 2.0 psig technical specification setpoint limit that initiates a high drywell pressure trip condition. However, only three minutes later (about 33 minutes after the initial scram and vessel isolation), drywell pressure increased sharply to about 2.7 psig.

Drywell-to-suppression chamber differential pressure also increased rapidly during this time. The jump in drywell pressure and drywell-to-torus differential pressure appeared to occur just when the "A" SRV lifted for the second time. From this evidence, operating personnel believe that reactor steam discharged directly into the drywell when the "A" SRV was manually opened for the second time at 4:49 a.m.. Licensee (Georgia Power Company) operating personnel at Hatch Unit 2 believe that following the first actuation of the "A" SRV, the associated tailpipe vacuum breaker stuck in an open or partially open position after the valve disk opened normally earlier in the event to prevent a vacuum buildup in the tailpipe. When the same "A" SRV lifted for a second time at 4:49, the (partially) stuck open vacuum breaker allowed steam to be released directly into the drywell, quickly raising drywell pressure. The vacuum breakers at Hatch Unit 2 do not incorporate position indication devices to assist the operator in positively determining the valve position.

After the unit was shut down personnel entered the drywell and inspected and tested all of the SRV tailpipe vacuum breakers. All of the vacuum breakers were found operable, and none showed a tendency to stick open when manually actuated. Additionally, none of the nearby equipment in the drywell showed any signs of steam impingement. GPC did not repair or replace any vacuum breakers prior to restarting the unit.

4.4 Loss of Drywell and Control Rod Drive Cooling

The drywell chiller units and the control rod drive pumps tripped off when drywell pressure exceeded the high pressure trip setpoint about 34 minutes after the reactor scrammed. When these systems tripped, normal drywell cooling and control rod drive cooling were lost. Although these actions occurred by design, as discussed in Section 2, they complicated post-scram recovery activities for the control room operating personnel. When a high drywell pressure (LOCA) signal occurs, the drywell chiller units and the CRD pump electric motors are tripped from the vital bus(es) since these equipment are not required for accident mitigation. Furthermore, since the drywell chiller units are not environmentally qualified for operation during an accident, stripping their electrical loads from the (emergency) bus(es) precludes a potential fault condition from feeding back to the emergency power supply during a postulated loss of coolant accident inside containment.

At Hatch Unit 2, after the CRD pumps trip on a LOCA signal it is possible to quickly restart the pump motors by bypassing the trip signal from a local control panel near the pumps. A handswitch is provided to switch operation of the pumps from automatic to manual control. With the switch in the manual position, the CRD pumps can be started with the accident signal still present. As discussed in Section 2, tripping the CRD pumps will cause the seals to heat up above the normal operating range. To possibly avoid this situation in the future, GPC is evaluating modifications to remove the CRD pump motors from the current load shedding arrangements.

At Hatch Unit 2, the drywell chiller units cannot be restarted as easily as the CRD pumps with an accident signal present. Bypassing the accident signal requires sending an electrical technician to the equipment cabinets to mechanically disconnect the proper lead for the trip circuit of the circuit breaker associated with the drywell chiller units. Given the lack of specific training and established procedures for this activity, this task generally takes a significant time to successfully complete. During the Hatch event, more than one hour and 45 minutes elapsed from the time the chillers tripped until the time they were brought back on line.

4.5 RCIC Isolation and Fire Suppression System Actuation

The cause for the RCIC isolation on high turbine exhaust diaphragm pressure was also investigated by GPC. Personnel entry into the RCIC NW corner room following the event revealed that an unintended opening existed in the normally closed embedded clean radwaste (CRW) drain piping. The opening would have enabled the hot scram exhaust water (which was believed to have been continuously discharging from the SDV system directly into reactor building equipment drainage system sump during the event) to backflow into the RCIC room through the interconnected drain line embedded in the floor of the basement. The hot steam which emanated from the drain opening wetted down and significantly increased the temperature of the electrical equipment and mechanical devices located in the room. The temperature increase was sufficient to set off the fire suppression system sprinkler head located immediately above the CRW drain system opening. The "tell-tale" on the RCIC turbine oil temperature gauge indicated the oil temperature had reached approximately 180°F. This is significantly above the normal 100°F operating temperature. Paint on the wall directly above the open CRW drain hub was also found to be blistered and peeling from the intensity of the heat.

Calibration tests were performed on the RCIC room equipment following the event. The tests revealed that all of the electro-mechanical instruments had drifted. None of the instruments were still within their permissible setpoint tolerance bands. The calibration tests also determined that the elevated room temperature had caused the trip setting for the Barksdale pressure switch, used for the RCIC turbine exhaust diaphragm high pressure isolation function, to drift down from 8 psig to 0 psig. That is, the switch contacts closed as a result of temperature-induced mechanical deflection during the event, initiating the spurious RCIC turbine exhaust diaphragm high pressure isolation signal which occurred at 5:10 a.m.

As a result of the steam release, all of the electrical equipment in the RCIC room was examined, dried, tested and recalibrated as required. Additionally, the entire electrical portion of the RCIC controller was removed and replaced with new components. In the long term, a previously planned analog trip system incorporating transmitters and bistables will be installed to replace the mechanical switches and trip devices used in the current instrumentation and control system. It is expected that this change will substantially reduce the setpoint drift associated with changes in the ambient room conditions.

As indicated previously, steam was released to the NW corner (RCIC) room during the event through an opening in the CRW drain piping embedded in the reactor building basement concrete floor. As discussed in Section 3.2, the CRW drainage system is designed as a closed piping network. However, plant personnel who entered the RCIC room observed that the threaded stainless steel pipe cap was missing from one of the short drain hubs. The unintentional opening in the otherwise closed CRW piping network permitted steam to be released into the room during the event.

Steam vapor also was reported to have leaked out of the reactor building equipment drain sump and the adjacent connected reactor building floor drain sump during the event. Both of these drain sumps are located in the SW corner room. It is believed that steam escaped from the sumps through either the various leakage paths associated with the covers and penetrations on the tops of the sumps and/or the connected local SW corner room instrumented floor drain sump.

Georgia Power Company representatives believed that the missing cover was removed from the drain hub in the NW corner room several months before the event. Although not certain, they believed that this was done in order to provide a temporary access to the drain system which was needed for RCIC room equipment testing (Ref. 5). Following the completion of these activities it is believed that plant personnel failed to put the cap back in place. Additionally, some time later but before the event occurred, plant personnel noticed that the cap was missing. Plant personnel informally requested that a replacement cap be installed on the open hub. No corrective actions were taken prior to the event, however.

As an immediate corrective action, following the event, a replacement stainless steel cover was screwed back on to the top of the open drain hub. Additionally, GPC representatives indicated that controls over the CRW drain hub caps will be upgraded for the future (Ref. 5). The caps will be tack welded in place and a specific maintenance authorization will be required to break the weld to remove the caps. The maintenance procedural controls involved will specifically address the need to replace the covers following completion of the equipment related activities requiring their removal.

4.6 Scram Discharge Volume Drain Valve Failure

The cause for the steady inflow of hot fluid into the reactor building equipment drainage sump was also investigated by GPC. During the event operating personnel thought that its source was the pressurized SDV. The SDV is located on the 130' elevation (floor) of the reactor building which is about 43' above the equipment drain sump basement elevation. The vents and drains from the SDV are hard-piped to the embedded clean radwaste drain system piping (refer to Section 3.2). The SDV is normally automatically isolated from the CRW drain system piping during a reactor scram by air operated isolation valves installed on the SDV system vent and drain lines. With this arrangement, the reactor water which is exhausted through the scram outlet valves during a reactor scram should normally be contained within the SDV system exhaust headers.

Scram discharge volume equipment testing was conducted by plant personnel following the event. The tests revealed that the isolation valve installed on the common drain line for the SDV headers would not fully close when actuated by its air operator. Upon closer examination, plant personnel observed that the operator yoke was loose from the valve body. This was caused by loose valve body-to-operator yoke retaining nuts which secure the yoke to the valve body. With a loose yoke the air operator was able to push away from the valve body when the valve operator attempted to stroke the valve closed during the event. Thus, tight seating of the valve plug could not be achieved when the valve received a close signal. An examination of the internals of the disassembled valve revealed no unusual material degradation or component sticking problems. The valve and its internals were cleaned, the air operator (yoke) was tightly secured to the valve body, and the valve was reinstalled on the drain line.

5. EVENT ANALYSIS AND EVALUATION

This section provides an analysis and evaluation of the Hatch Unit 2 event. The assessment provided here is divided into two parts. The first part contained in Section 5.1 discusses the safety significance of the event sequence from an overall integrated viewpoint. The second part presented in Section 5.2 provides a more detailed analysis and evaluation of the specific systems and components which had a significant involvement in determining the overall sequence of events.

5.1 Composite Event Sequence

5.1.1 Mitigation of the Limiting Event

The event that occurred at Hatch Unit 2 on August 25, 1982 involved several important elements of a postulated BWR accident scenario described in the USNRC report NUREG-0785 (Ref. 9). The report, entitled "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," describes a postulated event sequence in which, following a reactor scram at a BWR facility, a pipe fails in the SDV system. In the postulated scenario, the leak cannot be terminated immediately due to the presence of a reactor trip condition which cannot be reset. As a result, with the reactor still pressurized, hot reactor coolant discharges outside of the primary containment structure and into the secondary containment (reactor building) for an indefinite period of time. The eventual concern, discussed in NUREG-0785, is that continued release of high temperature reactor coolant could threaten the standby safety systems needed to assure safe shutdown. These systems are located in the reactor building. The challenge to the vital equipment is caused by the adverse environment postulated to develop in the reactor building. This environment, which includes possible flooding, could exceed the conditions for which the equipment is qualified. As a result, the conditions created by the accident (i.e., the break) might disable the equipment needed to mitigate the accident.

The loss of integrity to the SDV that occurred at Hatch Unit 2 following the reactor scram on August 25, 1982 was caused by neither a crack nor a break in the system piping. The loss of integrity resulted instead from incomplete closure of an installed drain line isolation valve. Even so, the partially stuck open valve would not have caused a significant plant problem in the long term except for the presence of a high drywell pressure scram signal which could neither be bypassed nor cleared. The high drywell pressure condition prevented scram reset for several hours. Thus, it was not possible for the operators to quickly terminate the leakage of hot reactor coolant into the reactor building. As a result, reactor coolant blew down outside primary containment into the reactor building clean radwaste drain system. Even so, the reactor coolant inventory which was lost could have been contained within the normally "closed" clean radwaste drain system except for a missing equipment drain hub cover in the RCIC room and leakage from the reactor building equipment drain sump. These pathways allowed high temperature fluids to be released into the surrounding open spaces of the reactor building.

As discussed in NUREG-0785, the release of high temperature reactor coolant directly into the reactor building potentially could threaten the operability of the standby systems which might be used to mitigate the event. This actually occurred during the Hatch event when the RCIC system isolated on a spurious isolation signal while injecting into the vessel to control level. The spurious isolation signal was caused by the adverse steam environment. In addition, the 130°F ambient temperature, which was reported to have been attained on the floor above the RCIC room, exceeded the qualification temperature for various vital motor control centers and panel boards located there. Included among the motor control centers were those associated with the valve motor operators for the low pressure emergency core cooling systems. However, no problems with vital equipment performance were reported as a result of the elevated temperature environment.

The postulated scenario and the associated safety concerns discussed in NUREG-0785 have been comprehensively studied and evaluated in detail on a generic basis by the General Electric Company (Ref. 10) and the NRC staff (Ref. 11). The postulated accident reviewed by the staff, and documented in NUREG-0803, involved a leakage crack in the SDV header. The size of the crack considered results in a flow area effectively equivalent to the full cross sectional area of the SDV drain line. The leakage crack considered by the staff, would result in consequences significantly more severe than the event that occurred at Hatch Unit 2 on August 25, 1982.

As a result of their review, the staff formulated a series of additional positions and guidance which were considered sufficient to assure that a BWR plant such as Hatch Unit 2 could adequately mitigate the postulated accident. For example, a guidance item in Reference 11 states that BWR plant emergency procedures should be revised to direct the operator to manually initiate a prompt emergency reactor pressure vessel (RPV) depressurization "whenever a trip condition that cannot be reset occurs coincident with indication of a leak in the reactor building or a leak that cannot otherwise be isolated." The purpose of this guidance is to reduce the rate of blowdown from the reactor into the reactor building. This operator action would lessen the severity and duration of the adverse environment caused by a leak or a rupture of the SDV system following a reactor scram.

In response to this guidance, procedures are being added to the BWR Emergency Procedure Guidelines developed by the BWR owners (Ref. 12). The additional procedures direct the plant operator to initiate an emergency RPV depressurization if the primary system is discharging into an area and the area temperature exceeds its maximum safe operating value (Ref. 13). Reference 13 is still being reviewed and has not yet been approved for use in formulating new symptom-based plant-specific emergency procedures. Thus, the emergency procedures and operator training in effect at the time of the Hatch Unit 2 event were not yet rewritten to lead the operator to take actions to initiate a rapid and early reactor vessel depressurization. Accordingly, operating personnel did not take steps to quickly depressurize the reactor during the event. The reactor was not quickly depressurized, even though operating personnel believed the scram system was exhausting high temperature fluid at high pressure into the reactor building (CRW), while an RPS trip condition which could not be reset existed. Although not implemented in time for the Hatch event, when implemented, the supplemented emergency procedure guidelines should significantly improve operator actions taken in response to a similar event, should one occur in the future.

Additional guidance contained in Reference 11 addresses the environmental qualification of safety systems which are needed to either detect a break in the SDV piping or to depressurize and shut down the reactor. Reference 11 suggests that such equipment and systems should be qualified to perform their intended function in the adverse environment resulting from an otherwise unisolable SDV system leak in the reactor building. In response to this view, BWR licensees (including GPC for Hatch Unit 2) have submitted documents to the staff, which address the issue of equipment qualification. In addition, since NUREG-0803 was issued, the General Electric Company and a group of BWR owners have requested that the NRC staff reconsider the need for environmental qualification of the safety systems which would be relied upon to mitigate the consequences of a break in the SDV system. This request was supported by supplemental information which attempted to assess the likelihood of a break in the SDV system. These submittals have not yet been fully reviewed and accepted for all affected BWR plants, including Hatch Unit 2. When the NRC staff's equipment qualification review (of the need for equipment upgrades) is completed, adequate mitigation capability will have been addressed.

In summary, the Hatch Unit 2 event of August 25, 1982 may be viewed as a less severe "precursor" to a more limiting but postulated accident sequence that currently is undergoing final review by the NRC staff. The generic analyses and evaluations which have been provided to date by the NRC staff for the more limiting postulated sequence are also considered bounding for the Hatch Unit 2 event, including its underlying causes and consequences (Ref. 11). Furthermore, from their review, the staff has developed guidance which is intended to assure adequate mitigation capability for the more limiting postulated accident scenario. If implemented at the affected BWRs, this guidance would assure adequate mitigation for any future "Hatch-like" events. The guidance includes such areas as emergency procedures and environmental qualification of needed safety systems. Guidance for the former issue has been finalized and is currently being implemented while the guidance for the latter issue is still undergoing final staff evaluation. Regardless, it is expected that the staff's final position on equipment qualification will consider the implications of the Hatch event including its causes and consequences. Accordingly, and in view of the staff's ongoing evaluation of the more limiting event senario, no additional detailed analyses or evaluations of the overall Hatch Unit 2 event sequence or its actual consequence is considered necessary for presentation in this report. However, a detailed analysis and evaluation of the specific causes for some of the important contributors to the event sequence (including the specific equipment and procedural problems involved) will be addressed in Section 5.2.

5.1.2 Multiple Failures and Problems

An additional consideration relating to the actual integrated Hatch Unit 2 event sequence involves the total number of equipment problems that occurred. At least five separate independent primary failures occurred during this event. These resulted in four additional consequential or secondary occurrences. The primary equipment failures were: (1) spurious closure of the inboard MSIV on the "C" steamline, (2) inoperability of the "H" SRV, (3) leakage of the vacuum breaker on the "A" SRV tailpipe, (4) a partially stuck open SDV system drain valve, and (5) a missing cap on the RCIC room CRW drain hub. The consequential equipment problems included: (1) tripping of the drywell chiller units, (2) tripping of the CRD pumps, (3) isolation of the RCIC system, and (4) actuation of the RCIC room fire suppression system. Of the five primary failures, it was necessary for at least four (i.e., MSIV, SRV vacuum breaker, SDV drain valve, and CRW drain hub cover) to occur together in order for the most significant event consequence to occur (i.e., RCS blowdown into the reactor building). The absence of any one of these four failures would have prevented the release of reactor coolant into the reactor building from occurring. Thus, the event provides an important example of the inherent limitations associated with the application of the "worst single failure" assumption frequently used in the study of potentially serious events. Serious events which are outside the traditional design basis can also occur when several independent and apparently inconsequential failures occur together in the same event.

5.1.3 Safety-Related/Nonsafety-Related Systems Interactions

Several systems interactions occurred during the Hatch event which involved both safety-related (or important to safety) and nonsafety-related equipment. The interactions also generally resulted from a failed or significantly degraded system isolation device associated with a system boundary. For example, the RCIC system (a system important to safety), failed due to a spurious isolation of the RCIC steamline. This was caused by a systems-interaction with the nonsafety-related clean radwaste drainage system. The interaction of the CRW system with the RCIC system was permitted by the missing system isolation device - in this case, the drain hub cover - associated with the "closed" CRW drain system piping.

The CRW drainage system extends into most of the vital equipment areas of the reactor building at the Hatch plants, as it does at most other BWRs. As discussed in Section 3.3, the reactor building equipment drainage system, unlike the reactor building floor drainage system, does not utilize either automatic or manual power operated valves or check valves to isolate portions of the piping network from the rest of the drain system to prevent the spread of an adverse environment to different rooms in the building. Isolating the CRW drain system from interacting with vital equipment in the reactor building depends on the administrative control over potential drain system boundary openings (i.e., covers). This interaction underscores the common cause failure potential of the reactor building equipment drain system arising from degraded isolation devices (i.e., missing or degraded drain hub covers).

The nonsafety-related reactor building equipment drainage system was in turn acted upon by the safety-related SDV system. This occurred when an isolation device of the SDV system boundary failed to function properly (i.e., the SDV drain valve failed to fully close). This failure permitted the potentially degrading effects of the high energy reactor water contained within the SDV system to be channeled throughout the reactor building via the "closed" CRW drain system. In effect, the combined failures of the isolation devices of both systems (i.e., SDV and CRW) involved a serious combination of failures which allowed the hot RCS coolant to interact with equipment in the reactor building. The sustained high drywell pressure scram condition allowed this interaction to continue for several hours.

Another significant interaction between safety-related and nonsafety-related equipment involved the main steam system interacting with the primary containment system and eventually the reactor protection and electrical power systems. The interaction of the main steam system with containment system was permitted by a partially degraded system isolation device associated with the main steam system. In this case, it is believed that a partically stuck open SRV tailpipe vacuum breaker allowed steam from the main steam system to pass directly into the drywell air space.

When drywell pressure rose to the RPS scram setpoint, it also tripped the non-safety-related drywell chillers. Tripping the chillers on the same (accident) signal is intended to prevent a possible overload of the chiller fan motors (which are located inside containment) during a loss of coolant accident inside primary containment. Increased loads on the fan blades would be caused by the increased atmospheric (steam) density inside containment during an accident. To avoid a possible overload of the fan motors (and possibly their associated vital buses), power to the motor loads is interrupted on an accident signal. Tripping the chillers on the same (accident) signal, also disables the principal system which can be used to reduce drywell pressure below the high pressure trip setpoint following a nonaccident (i.e., transient) event, however. At Hatch Unit 2 the drywell chiller trip feature does not incorporate a convenient bypass arrangement which may be used to return the system to operation on an expedited basis. Except for the limited drywell purge capacity of the drywell ventilation system exhaust bypass lines or the drywell sprays, no convenient methods are readily available to bring drywell pressure back down below the high drywell pressure setpoint. Thus, at Hatch Unit 2 a high drywell pressure condition itself will effectively prevent operation of the principal system which can be used to clear a high drywell pressure condition once it occurs. This "Catch-22" systems interaction arrangement would normally be viewed as an acceptable anomaly of the design of the primary containment cooling system. However, for this event it was the principal cause for the delay in terminating the RCS blowdown outside of primary containment.

In summary then, this series of interaction caused and effectively prevented timely clearing of the high drywell pressure scram signal. Timely reset would have been necessary to quickly terminate the release of hot reactor water outside primary containment via the SDV volume.

5.2 Specific Systems and Equipment

This section provides a further assessment and additional information related to specific systems and equipment which played a significant role in the Hatch Unit 2 sequence of events.

5.2.1 Main Steam Isolation Valve Failure

As discussed in Section 4.1, the inboard isolation valve on the "C" main steam line closed unexpectedly when the main valve disk separated from the valve stem. This was caused by disengagement of the poppet from the stem. With the poppet off, the valve disk was free to drop off the stem. The valve which failed is a "Y" pattern globe valve manufactured by Rockwell International. There have been several other similar mechanical failures of the "Rockwell-Edward Flite Flow Stop Valve" at different BWRs, including a previous occurrence at Hatch Unit 2. On March 5, 1981, at Hatch Unit 2, the "A" steamline inboard isolation valve disk separated from the stem. At least seven of the MSIV failures including the earlier event at Hatch Unit 2, occurred between January 1976 and July 1981. The Brunswick facility reported almost all of the failures during the January 1976 to September 1981 period. In each case, the valve disk separated from the stem.

In September, 1981, the USNRC issued an Inspection and Enforcement (IE) information notice on this subject, to all power reactor facilities with an operating license or construction permit (Ref. 14). All of these events were covered by the IE information notice. In addition to Hatch Unit 2 and the two Brunswick plants, other BWRs, including Cooper, Duane Arnold, Fitzpatrick and Vermont Yankee use the Rockwell-International valve for the main steam isolation valves. Since the IE information notice was issued, at least three new similar MSIV failures have been reported, including the event at Hatch Unit 2 (Refs. 7, 15 and 16). The other two failures occurred at the Fitzpatrick plant in October and December of 1982. A review of the LER data base indicates that no additional failures have been reported at the Brunswick facility since the IE information notice was issued. This can likely be attributed to the special involvement by the valve manufacturer, Rockwell International (Ref. 17). Rockwell investigated the cause of the valve failures in order to develop recommended corrective actions needed to eliminate the valve problems at the Brunswick facility. Reference 17 describes three potential solutions to the disk-to-stem disassembly problem for the Rockwell valves. A review of the corrective actions described in the LERs for the Hatch Unit 2 event (Ref. 7) and the two Fitzpatrick events (Refs. 15 and 16) indicates that the corrective actions which were considered in Reference 17 for Brunswick have either not been finalized or have not been adequately evaluated and implemented at these other BWR facilities.

5.2.2 High Drywell Pressure and the Loss of Drywell Cooling

As discussed in Section 2 and Section 4.4, when pressure in the drywell exceeded the 2 psig high pressure trip setpoint the nonsafety-related drywell chiller units were tripped off-line because of load shedding logic associated with their (emergency) buses. As described in Section 4.4, the trip feature for the drywell chiller units is provided to prevent a potential faulted condition associated with the nonseismically qualified and nonenvironmentally qualified chiller equipment from adversely effecting the emergency power supplies during a postulated loss of coolant accident inside containment. Moreover, the chiller fan motors are generally sized to handle loads imparted on the fan blades by a fluid medium consisting of either air or nitrogen with relatively low moisture content. Following an accident, the predominantly gaseous fluid medium would be displaced by a fluid medium consisting of saturated steam and suspended water droplets. Thus, the density of the drywell atmosphere following an accident would be significantly greater than the drywell atmosphere density during normal plant operations. As a result, the fan motor loads would be expected to increase during an accident. To avoid overloading the fan motors and possibly the emergency buses during or following an accident, these motor loads are stripped from the bus. Thus, it may be concluded that tripping the drywell chiller units during an actual loss of coolant is a needed protective action.

However, as seen from the Hatch Unit 2 experience, actuation of the load shedding feature on high drywell pressure attendant to a transient may also lead to additional difficulties for the operating staff following a reactor scram and vessel isolation transient. This can occur if the transient is accompanied by sufficient steam leakage into the drywell to raise drywell pressure beyond the high pressure trip setpoint. When this occurs the principal system which would be used to reduce drywell temperature and pressure to clear the trip signal (i.e., the drywell cooling units) is also made inoperable by the high drywell pressure condition.

At Hatch Unit 2 and other plants, this systems interaction cannot be readily overcome. As a result, the normal activities associated with post-scram recovery operations are made more complicated. Furthermore, at Hatch Unit 2 no convenient arrangements are provided to quickly bypass the high drywell pressure signal to allow reclosure of the chiller unit trip breakers to quickly reestablish drywell cooling. There is no high drywell pressure override switch for the drywell coolers similar to the high drywell pressure override switches for the ventilation exhaust bypass lines. The plant operating staff at Hatch Unit 2 is trained on emergency bypassing of signals in general. However, at Hatch Unit 2 no specific pre-established emergency procedures or training have been provided to facilitate quickly locating and lifting the proper electrical leads for the trip circuit of the circuit breaker associated with the drywell chiller units. As a result of these arrangements, the difficulties caused by a lack of normal drywell cooling capability (i.e., the inability to clear the high drywell pressure scram condition) were substantially prolonged. It is interesting to note, that at Hatch Unit 1, a bypass switch is provided in the control room which allows the operator to quickly restart the chiller units with a high drywell pressure (LOCA) signal present (Ref. 30).

At least one other BWR has recently experienced similar difficulties associated with high drywell pressure following a reactor scram (Ref. 18). At approximately 5:25 a.m. on June 22, 1982, the Quad Cities Unit 2 reactor inadvertently tripped while operating at 95 percent power. Early in the event, reactor heat removal via the main condenser was lost. Reactor pressure increased to the opening setpoint of one of the relief valves which opened automatically. A second relief valve was manually opened to assist in controlling pressure in the reactor. At 5:55, or about 30 minutes after the reactor initially scrammed, drywell pressure reached 2 psig and initiated a second reactor scram signal, several containment isolations and the standby core cooling systems. It was later determined that the pressure increase was caused by leaking gaskets installed on both the main steam relief valve tailpipe flange connections and the blind flanges for tailpipe vacuum breaker valves that were to be installed at a later date. The gasket leakage allowed steam to discharge into the drywell when the main steam relief valves were actuated during the event.

The 2 psig drywell pressure signal also tripped the drywell cooler fan motors and the reactor building closed cooling water (RBCCW) system pump motors. At Quad Cities Unit 2, RBCCW supplies secondary side cooling to the drywell coolers. The steam released inside the drywell caused drywell pressure to increase to a maximum value of 4.3 psig. Because of the high drywell pressure and lack of drywell cooling an electrical technician also had to be dispatched into the reactor building to jumper out the high drywell pressure signal to the load shedding logic for the drywell cooler electrical supply. It was not until approximately 7:00 a.m. that drywell cooling with RBCCW flow to the coolers was reestablished. Reestablishing the coolers allowed the operators to begin to bring drywell pressure back down below 2 psig. Approximately one hour was required to reestablish drywell cooling from the time it was initially lost, however.

As a result of this event a change package was developed for the Quad Cities facility. The change modifies the emergency core cooling (core spray) initiation logic so that the drywell coolers and RBCCW pumps do not trip on a 2 psig drywell pressure signal if power remains available to the emergency buses. This change was based in part on supplemental plant specific studies which showed that the drywell cooler fan motors would not be overloaded when drywell atmospheric density increased following a postulated accident. With this change, the drywell coolers at Quad Cities will be available following a transient, even if drywell pressure increases above 2 psig. Thus the drywell coolers will remain operable to aid in controlling drywell pressure at times when the drywell cooling function is still needed. Similar supplemental analyses at other plants may not be able to show that the fan motors would not be overloaded, however.

5.2.3 Scram Discharge Volume Drain Valve Failure

A discussion of the SDV drain valve failure is provided in some detail in Section 4.6. As mentioned there, a loose valve body-to-operator yoke prevented the attached air operator from seating the valve plug tightly into its seat. When the drain valve failed to close fully during the prolonged high drywell pressure scram condition, hot pressurized reactor water escaped from the SDV headers. The escaping hot water and steam discharged directly into the reactor building equipment drainage system.

A similar event occurred at Brunswick Unit No. 1 on October 19, 1979 (Ref. 19). On that date a reactor scram occurred from full power and was caused by a spurious main steam line high radiation signal. Following the reactor trip, the SDV drain valve did not close for about 4 to 5 minutes. The normal valve closing time is approximately 30 seconds. The delayed closing of the drain valve was traced to a faulty three-way solenoid valve controlling the supply of air to the drain valve air operators. The faulty solenoid valve caused air to be bled off the air operator too slowly when the scram signal was received. As a result of the delayed closing time, hot pressurized reactor water discharged into the reactor building equipment drain system piping for several minutes. Damage to various equipment was sustained because of the high pressure reactor water which discharged past the stuck open valve during the event.

Prompted by investigations of the Brunswick event and the June 28, 1980 Browns Ferry Unit 3 partial scram failure event, the NRC staff determined that improvements would be needed in the reliability of the isolation arrangements of the SDV vent and drain lines. As a first step, on July 7, 1980 the NRC staff requested all operating BWR licensees to propose technical specification surveillance requirements for the existing SDV vent and drain valves (Ref. 20). The surveillance requirements were intended to be an interim measure which would assure SDV vent and drain valve operability on a continuing basis during reactor operation. The interim technical specifications were intended to provide adequate assurance that the existing SDV isolation valves would perform their intended function until such time that more extensive permanent modifications to the SDV system isolation arrangements could be completed.

To assist licensees in preparing their submittals the staff enclosed model technical specifications, with their request. The suggested technical specifications, which were considered sufficient to provide the assurance sought, involved additions to the Control Rod Operability section of the Standard Technical Specifications. The changes required that the SDV vent and drain valves be proven operable whenever the control rods were scram tested. This could be met by verifying that the vent and drain valves: (1) closed within a predetermined number of seconds after receipt of the scram insertion signal for the control rods and (2) opened when the scram signal was reset or the SDV trip was bypassed.

By letter dated February 26, 1981, GPC responded to the request by proposing changes to the Hatch Unit 2 technical specifications (Ref. 21). Georgia Power Company proposed to add the SDV vent and drain valves to the already existing tables for containment isolation valves. This change would have included the SDV isolation valves along with the normal surveillance requirements for these valves. The staff could not approve the technical specifications proposed by the licensee because they did not meet the staff position on surveillance testing for SDV vent and drain line isolation valves. The staff position required more stringent and more frequent testing than that which is normally associated with containment isolation valves. Accordingly, by letter dated September 1, 1981, the staff formally requested that additional information be submitted for staff review (Ref. 22). Georgia Power Company responded to this request in a letter dated October 1, 1981 (Ref. 23). The letter reaffirmed their position that the valves should be associated with assuring primary containment boundary integrity rather than assuring control rod operability. GPC provided neither a reference nor a basis for the proposed technical specification section modification involved in their February 26, 1981 submittal, as originally requested by the NRC staff. However, in a subsequent January 13, 1982 Technical Evaluation Report (TER), which was prepared by an NRC consultant for the Hatch Unit 2 SDV modifications, it was noted that the licensee had orally agreed to revise the proposed specification changes so that the valves would be surveillance tested in accordance with the staff requirements (Ref. 31). While the TER stated that GPC had informally agreed to meet the staff requirements for the surveillances, the licensee failed to agree in writing and continued to delay proposing the desired modified specifications. By the time of the August 25, 1982 event at Hatch Unit 2, and over two years after initially requested, acceptable technical specification surveillance requirements for the SDV system vent and drain valves had not been reviewed and approved by the staff. Had the required specifications been in place and implemented prior to the date of the event, it is likely that the SDV valve failure would have been avoided.

Following the Browns Ferry Unit 3 scram system failure, the NRC staff determined in its safety evaluation for the BWR SDV system that long term hardware improvements in the isolation arrangements for the SDV system would also be required (Ref. 24). Included in the safety evaluation report is a safety criterion which states that no single active failure shall prevent uncontrolled loss of reactor coolant. The staff noted that the SDV vent and drain lines at BWRs (including Hatch Unit 2) are normally equipped with a single isolation valve and that the failure of either (vent or drain) valve could result in an uncontrolled loss of reactor coolant following a reactor scram. It was the staff's position that the safety criterion was necessary to meet the "single failure" rule with regard to containment of reactor coolant. The staff noted that an acceptable way of meeting the criterion would be to provide two isolation valves in series in all SDV vent and drain lines. The valves would also have to be sufficiently independent in their operating arrangements to avoid the potential for common cause failure of both valves.

The staff requirement for redundant isolation valves for the SDV system piping, together with associated technical specifications for their operation, are considered acceptable permanent corrective actions for the SDV drain valve failure at Hatch Unit 2 on August 25, 1982. However, by the time of the event, neither corrective measure had actually been implemented at Hatch Unit 2. On June 24, 1983, a Confirmatory Order was issued to GPC for Hatch Unit 2 which confirmed the licensee's commitment to make the permanent SDV system modifications (including redundant vent and drain valves) by December 31, 1983. It also enclosed proposed model technical specifications for operating the plant with the modified system. The required modified technical specifications were finally approved for use by the staff in a letter to GPC dated January 4, 1984.

5.2.4 Missing Clean Radwaste Drain Hub Cover

As discussed in Section 4.5, hot reactor water and steam escaped from the reactor building clean radwaste drain system because of a missing RCIC room equipment drain hub cover. It is likely that the cover was removed several months earlier during RCIC room equipment maintenance or testing activities. According to Hatch plant personnel, covers are frequently removed from the drain hubs to provide needed access to the drain system for equipment hydro-testing, local leak rate testing, instrument surveillance testing and equipment calibrations. During these activities a tygon tube may be routed to a nearby open drain hub to collect and remove equipment leakoffs. Neither equipment maintenance nor equipment test procedures at Hatch Unit 2 specified replacement of the cover after removal following completion of the activity requiring its removal. At Hatch Unit 2, only general housekeeping instructions addressed the need to return important equipment (such as the equipment drain system) to its original condition or configuration after completing an activity. Additionally, routine observation of the missing cap prior to the event by the assigned system engineer resulted in a subsequent verbal request to have a replacement cap installed. However, a replacement drain hub cover was not provided prior to the event on August 25, 1982. Thus the administrative control arrangements were apparently inadequate to assure replacement of the drain hub cover.

Covers are frequently removed to perform routine equipment tests and maintenance activities within the reactor building and numerous drain hubs are located throughout the reactor building. Therefore, as discussed in Section 4.5, GPC is taking steps to strengthen the administrative controls over the drain hub covers. This change will reduce the likelihood of a drain hub cover being inadvertently left off in the future.

A similar problem involving equipment drain hub covers occurred at the Pilgrim Nuclear Power Station on November 15, 1982 (Ref. 25). On that date approximately 12 inches of water was found to have collected on the floor of the RHR system corner room and the HPCI room. The reactor building equipment drainage sump, which is located in the HPCI room at the Pilgrim plant, was found overflowing into a nearby floor drain. An investigation of the situation conducted by the licensee determined that water was overflowing from the reactor building equipment drainage sump due to an interconnected condensate demineralizer that was operating in the backwash cycle. It was also determined that the mechanism that allowed interaction between the sump and the RHR quadrant resulted from a prior modification to the equipment drainage system. The modification involved the RHR pump equipment drain and resulted in the nearby equipment drain standpipe being cut off at floor level and a cap epoxied in place. The licensee determined that the cap had become loose, allowing water to back up from the reactor building equipment drain sump into the RHR quadrant. To temporarily correct the problem, the licensee implemented an interim modification involving installation of an expanding plug to prevent inadvertent backup of liquid into the RHR corner room.

The importance of reactor building equipment drain hub covers involved in the recent Hatch Unit 2 and Pilgrim events was first brought to the attention of BWR licensees in a USNRC Inspection and Enforcement circular issued about four years earlier (Ref. 26). IE Circular No. 78-06, "Potential Common Mode Flooding of ECCS Equipment Rooms at BWR Facilities," dated May 25, 1978, describes concerns involving the arrangement of the reactor building equipment drainage system at BWR facilities.

The concerns expressed in the circular were based on a letter report from GPC following a design review of the reactor building equipment drainage system at Hatch Unit 1. GPC reported at the time that the various equipment drain lines at Hatch Unit 1 were piped into the top of a 6" open ended pipe (i.e., a standpipe) in each of the corner rooms (Ref. 27). The open standpipes in the corner rooms were in turn all cross-connected by the embedded reactor building equipment drainage system piping network. Furthermore, all of the standpipes fed to the common equipment drain sump located in one of the reactor building basement corner rooms. One of the reasons for installing the ECCS equipment in separate watertight rooms was to eliminate the potential for common flooding. GPC noted that the standpipe openings associated with the embedded equipment drain piping could subject the emergency core cooling equipment to common flooding, however. Accordingly, the circular explained that GPC was "hard-piping" all of the equipment drain lines into the side of the corner room standpipes and capping the open end of the standpipes with a removable cover. This arrangement would make the drain system a closed system as originally intended and as described in the FSAR.

The circular requested that operating reactor licensees investigate whether or not similar pathways which could lead to common flooding of redundant safety equipment existed at their facilities. Also, the circular recommended that administrative controls be reviewed to assure that separation criteria are maintained and watertight room separation devices such as doors and hatches are closed as appropriate.

The circular did not specifically mention the need for adequate administrative control over the reactor building equipment drainage system drain hub covers, although these would appear to be a clear example of the kind of separation devices which would be involved. Needed maintenance of the separation criteria (equipment) clearly should also encompass replacement (i.e., the event at Hatch 2) and/or repair (i.e., the event at Pilgrim) of the equipment drain hub covers. Thus, it would appear from the recent experiences at two different BWRs, that the administrative controls (relating to the maintenance of equipment drain hub cover separation devices) mentioned in IE Circular 78-06, have not been adequately implemented at all of the potentially affected plants.

5.2.5 Reactor Building Floor Drainage System

The reactor building floor drainage system at Hatch Unit 2 (refer to Section 3.3), came into use during the August 25, 1982 event. Liquid inflow into the system entered via open floor drains in the RCIC corner room. The water sources that entered the open floor drains consisted of the steam and hot water discharged from the open equipment drain hub plus the water sprayed from the actuated sprinkler head of the fire suppression system. Water collected by the RCIC room floor drain was delivered to the floor drain sump located in the SW corner of the reactor building basement. During the event, water did not accumulate on the floor at a sufficiently rapid rate to cause floor drains to isolate the RCIC room from the rest of the reactor building floor drainage system. This likely would have occurred had the rate of water flow into the room been greater than the rate at which the water could be removed by gravity flow to the reactor building floor drain sump. In such a situation, potential flooding of the other basement rooms would have been prevented by automatic closure of the drain line isolation valves located in pits in the concrete floor. This action would have been taken by design at the expense of possible worse flooding in the RCIC room.

At Hatch Unit 2, the open instrumented floor drain sump in the HPCI room and the open instrumented floor drain sump in the adjacent SE corner (RHR and CS) room empty into the same reactor building floor drainage sump in the SE corner room of the reactor building (see Figure 3 of Section 3). These sumps are cross connected via 6" diameter piping embedded in the basement floor. The instrumented floor drain sump in the SE corner room does not incorporate isolation valves to isolate it from the main large sump in the same room. The HPCI room instrumented floor drainage sump incorporates a single valve to isolate it from the adjacent SE corner room sump in the event of flooding in either room.

Although the reactor building floor drainage system incorporates adequate protection against common mode flooding of the separated vital areas in the basement, the installed equipment may not be adequate to isolate vital equipment areas from the harsh environment resulting from high energy line breaks in these areas. The Final Safety Analysis Report for Hatch Unit No. 1 states that the peak pressure attained in the HPCI room is 26.6 psi following the largest HPCI steam line break located in the room. The analysis also indicates that 63.0 seconds is required to fully isolate the break from the time the accident initially occurs. This assumes normal ac power is lost and the worst single failure is the dc operated HPCI steam line isolation valve which fails to close. For this postulated sequence, 13 seconds are required to reestablish onsite ac power to the operable (ac) valve, and another 50 seconds are required to close the valve. Assuming no failures and no loss of offsite power, 50 seconds are required to close the two isolation valves.

The transient pressure in the HPCI room would be applied at all points in the room including the instrumented floor drain sump. The steam pressure in the room would be expected to cause some of the superheated steam to enter the instrumented floor drain sump. Once there, the steam would be channeled by the embedded piping to the reactor building floor drain sump in the adjacent corner (RHR and CS) room. Steam would continue to flow through the embedded floor drain piping until insufficient driving pressure was available in the HPCI room or the HPCI room floor drain sump was automatically or manually isolated from the adjacent reactor building drain sump. Since very little liquid water would actually enter the reactor building sumps during the accident to cause a sump level rise, it is unlikely that the HPCI room instrumented drain sump would isolate automatically by actuation of the level switch associated with the reactor building floor drain sump located in the SE corner room. Thus, it could be expected that steam would also be released to the SE corner room via either the top of the reactor building floor drain sump and/or the connected corner room floor drain sump during a HPCI steam line blowdown into the HPCI room. The backflow mechanism involved would be analogous to the mechanism involved in the steam release to the RCIC room during the August 25, 1982 event at Hatch Unit 2. In this case, however, the floor drain system rather than the equipment drain system would be involved.

As shown in Figure 3, at Hatch Unit 2 the ECCS equipment in the NE corner room is also connected to the HPCI room and the SE corner room via the normally open floor drain system. However, it should be expected that the flow resistance associated with the length of the piping run to the more distant NE corner room would prevent significant steam back flow in to this area.

The quantity and properties of steam entering the SE corner (RHR and CS) room from the HPCI room would determine the environmental conditions that would develop in the room. If the environmental conditions exceeded the design basis for the equipment in the room, operation of the equipment would not be assured. Thus, a break in the HPCI room might result in the consequential loss of one of the two divisions of low pressure ECCSs that are normally provided to mitigate the accident. This might occur as a result of the adverse environment being channeled to the nearby room through the open and unisolated floor drain system. Furthermore, the HPCI steam line break analysis normally assumes a concurrent loss of off-site power and a worst single failure. If the worst single independent failure is taken to be a starting failure of the diesel generator for the divisional ECCS equipment in the NE corner room, then less ECCS mitigation equipment would likely be available than that which has been assumed heretofore in the plant safety analysis.

5.2.6 Safety /Relief Valve Tailpipe Vacuum Breaker Failures

As stated previously, GPC was not able to find direct evidence that the drywell pressure increase, which occurred during the event, was caused by a stuck open SRV tailpipe vacuum breaker. Even though no hard evidence could be developed for this event, a recently completed study of BWR SRV tailpipe vacuum breaker operating experience found that breaker failures have occurred previously at Hatch Unit 2 (Ref. 28). The study discusses five vacuum breakers which were found partially stuck open on March 9, 1979 at Hatch Unit 2. The vacuum breakers which were involved incurred damage to the disk hinge pin and spring. More recently, on December 15, 1983, GPC reported damage in a licensing event report to two of the SRV tailpipe vacuum breakers at Hatch Unit 1 (Ref. 29). Reference 28 also describes SRV vacuum breaker problems which occurred at Peach Bottom 2 and Browns Ferry 1. A total of 14 damaged vacuum breakers were found in the four events. Reference 28 recommended that a review of the adequacy of current criteria for the design and installation of the vacuum relief devices be undertaken. The study conclusions also indicate that the USNRC's Office of Nuclear Reactor Regulation had initiated such a review.

6. FINDINGS AND CONCLUSIONS

The plant systems interaction transient that occurred at the Edwin I. Hatch Unit 2 Nuclear Plant on August 25, 1982 was significant in terms of both the underlying causes and the resulting plant systems response which were involved. Individually, the contributing equipment failures and problems that occurred were of limited safety significance. However, the combined effects of the failures and problems resulted in a very significant consequence - a sustained uncontrolled and unisolable blowdown of the reactor coolant system outside primary containment. The event discussed in this report would appear to mark the first time that a prolonged uncontrolled reactor blowdown has occurred outside of containment from hot pressurized conditions at a domestic commercial BWR nuclear power plant. The event represented a serious and simultaneous degradation of both the reactor coolant pressure retaining boundary and the primary containment boundary.

The locally harsh environment which resulted from the discharge of high energy fluid was sufficiently severe to cause the operating standby high pressure make up system (i.e., the RCIC system) to shut down while it was injecting into the isolated reactor coolant system. The peak ambient temperature in other parts of the reactor building some distance from the point of discharge was reported to be sufficiently high that it likely exceeded the environmental qualification temperature for the safety-related electrical equipment located there. Thus, the blowdown had the potential to threaten the operation of vital equipment in other areas of the reactor building some distance from the discharge source. However, at no time during the event did a significant potential exist for inadequate core cooling or inadequate core cooling capability.

The prolonged RCS blowdown followed a reactor scram from full reactor power, which was initiated when one of the MSIVs failed closed unexpectedly. The reactor trip was accompanied by a vessel isolation from a high steam flow condition in the steam lines, which remained open. The HPCI and RCIC systems successfully auto-started to maintain water level. Several minutes after the first scram a high drywell pressure scram condition occurred which could only be reset by the plant operating staff several hours later. The coolant lost from the reactor passed through the primary containment and flowed through the control rod drives to the SDV system located in the reactor building (secondary containment). The reactor water lost from the system exited through a partially stuck open drain line isolation valve downstream of the open scram outlet valves and SDV headers. The loss of inventory could not be quickly terminated due to the sustained high drywell pressure scram signal. The scram signal, which could not be bypassed, prevented a routine and early reclosure of the upstream scram outlet valves via reset of the reactor protection system. The inventory lost might have been effectively contained within the closed reactor building drainage system sumps. However, a normally installed equipment drain hub cover was missing in the RCIC corner diagonal room. The drain system piping back-channeled the discharging hot reactor water from the sump to the RCIC room open drain hub to be released into the reactor building. The steam and hot water released created a sufficiently harsh environment to shut down the operating RCIC system and set off the fire suppression system.

The Hatch event may be viewed as a "precursor" to a similar but more limiting "postulated" accident sequence that has recently been comprehensively reviewed by the NRC staff on a generic basis. In the postulated accident sequence the coolant lost from the reactor was assumed to exit the SDV system directly into the open areas of the reactor building through a break in the SDV drain line. If implemented, the staff positions and guidance which followed the staff's review of the more limiting postulated accident would assure that adequate mitigation capability would be available for even the more limiting accident case. The guidance includes such areas as emergency procedures and environmental qualification of needed mitigation systems. Guidance for the former issue has been finalized and is currently being implemented. Guidance relating to the need for environmental qualification of the needed safety systems is undergoing final staff evaluation. Nevertheless, the Hatch experience provides a clear example of the potential for the BWR SDV system to cause an unisolable RCS blowdown outside containment. Previously, this potential had only been postulated for purposes of analysis and evaluation of the consequences and for determining the mitigation requirements for such an event.

The underlying causes of the equipment failures and problems which occurred during the Hatch event are also significant in that most were addressed in NRC correspondence to GPC (and other BWR licensees) well before the event occurred. The transmittal of these documents was prompted by the NRC staff's review of prior operational experiences at the Hatch plant and other nuclear power reactor facilities. The communications contained substantial information relating to the failure causes and the needed corrective actions.

The main steam isolation valve failure which initiated the plant transient was addressed in an IE information notice issued in 1981, about a year before the event. The notice described the causes for prior similar MSIV failures and possible corrective actions. It may be concluded that the lessons learned from these earlier MSIV failures were not adequately assimilated and acted upon by GPC prior to the event. The SDV drain valve failure that occurred during the event also might have been avoided had GPC been able toimplement proposed technical specification surveillance requirements for the SDV vent and drain line isolation valves. That is, as early as July 1980, GPC was requested in a generic NRC letter to add these valves to the existing technical specification surveillance testing requirements by proposing technical specification changes which would accomplish this purpose. This request was based on the NRC staff's evaluation of the BWR SDV system following a partial scram failure event at the Browns Ferry plant. Approval and implementation of the requested technical specification changes was finally completed in January 1984.

Similarly, the need for adequate administrative control over devices such as the reactor building equipment drain hub covers was brought to the attention of Georgia Power Company and other BWR licensees in an IE circular over four years before the event. The circular was prompted by and based upon the experience gained from a design review of reactor building equipment drain hub openings at the Hatch facility. The circular recommended that adequate administrative control of such separation devices should be provided. However, it may be concluded that the corrective actions taken in response to the circular were not adequate to prevent a cover from being left off the drain hub in the RCIC room for a considerable period before the event.

In summary it would appear that the sustained blowdown of reactor coolant into the secondary containment at Hatch 2 on August 25, 1982 could have been avoided had any one of a number of equipment problems been prevented. It may also be concluded that most of these equipment problems could have been prevented had the lessons learned from previous operating experience been adequately implemented in a timely manner.

Finally, the Hatch event underscores the potential of the reactor building equipment and floor drainage systems to channel adverse environments, including flooding, to remote areas of the reactor building via their interconnected piping networks. In this regard the floor drain system may also have the potential to channel the harsh environment associated with high energy line breaks (outside containment) to vital areas of the reactor building which are otherwise protected against such harmful conditions.

7. RECOMMENDATIONS AND SUGGESTIONS

(1) It is suggested that an industry representative or group, such as INPO, take appropriate actions to alert potentially affected BWR operators of the continuing stem-to-disk separation problems associated with Rockwell-Edward "Y" pattern globe-type main steam isolation valves. It is further suggested that an industry representative or group identify and feedback to the BWR operators the corrective measures needed to prevent the recurrence of disk-to-stem separation of Rockwell-Edward MSIVs.

IE Information Notice 81-28, "Failure of Rockwell-Edward Main Steam Isolation Valves," which was issued on September 3, 1981, informed all BWR licensees of the disk-to-stem separation mechanical failures reported for Rockwell-Edward MSIVs. Up to the time the notice was issued, almost all of the failures which were reported occurred at the Brunswick Unit 2 facility. One separation failure mentioned in the IE notice occurred at Hatch Unit 2. Since the notice was issued, three additional failures have been reported. Two have occurred at Fitzpatrick and one more at Hatch Unit 2. This second Hatch event occurred on August 25, 1982 and initiated the plant transient discussed in this report. No additional failures have been reported for the Brunswick facility since the information notice was issued. A review of the "Corrective Actions" section of the LERs for the most recent failures reported at Fitzpatrick and Hatch Unit 2 indicates that the corrective measures developed and taken earlier at the Brunswick plant were not being utilized at these and possibly other potentially affected BWR plants. Accordingly, we would propose that an industry representative or group, such as INPO, identify and feedback to BWR operators the corrective measures needed to prevent the recurrence of disk-to-stem separation of Rockwell-Edward MSIVs. At the same time we have no reason to believe that the subject stem-to-disk separation phenomena can prevent closure of the MSIVs, which would represent a much more significant failure mode were it possible.

(2) It is suggested that the Office of Inspection and Enforcement take appropriate actions to follow-up IE Circular No. 78-06, "Potential Common Mode Flooding of ECCS Equipment at BWR Facilities," to ensure that operating reactor licensees are providing adequate administrative and/or physical controls to maintain of the required condition and placement of reactor building equipment drain hub covers.

Inspection and Enforcement Circular No. 78-06 described a common flooding potential which could be caused by open pipes (standpipes) of BWR reactor building equipment drainage systems. The circular provided the corrective actions taken at the Hatch facility where the problem was first identified. The changes made at Hatch involved hard piping all of the equipment drains into the side of the standpipes and capping the open tops of the standpipes. The circular recommended that licensees consider reviewing their facility drainage arrangements for a similar flooding potential. The circular also recommended that licensees review the adequacy of their facility administrative controls involving the maintenance of vital equipment room separation devices (e.g., hatches and doors).

Since the circular was issued, at least two recent events have been found which involved the spread of a potentially harmful environment to separated vital areas in the reactor building due to an improperly sealed equipment drain opening. In both cases, the loss of drain system integrity was traced to a failure to maintain the original sealing arrangements provided in response to the circular for the drain system opening. As a result of these events, it is suggested that IE take appropriate steps to verify that adequate administrative and/or physical controls are being provided by licensees to ensure proper condiition and placement of areactor building equipment drainage system (hub) covers. Such verification could include a Temporary Instruction written to resident inspectors to follow-up on the actions taken by licensees in response to Circular 78-06.

(3) It is recommended that the Office of Nuclear Reactor Regulation assess the extent to which separated and protected nearby mitigation systems might be consequently effected (degraded) during a steam line break accident outside containment as a result of portions of the released steam being channeled within the reactor building floor drain system. If the results of such an assessment reveal that the local environment exceeds the design basis for equipment needed to mitigate the accident or safely shut down the plant, it is further recommended that supplemental arrangements be provided to assure timely isolation of the affected drainage system piping network.

The Hatch event demonstrated that embedded drain systems can back-channel adverse environments to distant areas of the reactor building from their point of origin. In this way, standby emergency equipment can be disabled by the resulting harsh environment even if no flooding threat exists. The open floor drain system is equipped with an automatic isolation capability designed to function in and protect against a rising liquid level situation (i.e., flooding). In the event of a high energy line break in the reactor building (e.g., an HPCI steam line break in the HPCI room), the high compartment pressure could force steam through the floor drains to other nearby compartments. At the same time, the low (steam) density would not necessarily cause a sump liquid level rise needed to actuate the isolation equipment. The consequential safety system failure(s) which may result might reduce the mitigation capability below an acceptable level when taken together with the usual single random failures considered in the overall accident analysis. The evaluation recommended could be included in the resolution of Potential Generic Issue No. 77, "Flooding of Safety Equipment Compartments by Back Flow Through Floor Drains."

(4) <u>It is suggested that the Office of Inspection and Enforcement consider issuing an information notice to BWR licensees describing the recent drywell pressure events at Hatch Unit 2 and Quad Cities Unit 2*</u>

In both the Hatch and Quad Cities events (refer to Section 5.2.2), high pressure in the drywell following a reactor scram resulted in a loss of normal drywell cooling due to load shedding logic. The loss of normal drywell cooling prevented the operators from readily reducing drywell pressure below the accident initiation setpoint required to restart the cooling units. In both events an electrical technician had to be dispatched into the reactor building to disconnect electrical leads from the trip circuit of the circuit breaker for the cooling units. This response was necessary since no hand switches were provided to override the high drywell presure signal in a nonaccident situation. For both events, lengthly time delay was required to return the cooling equipment to operation and to subsequently lower the actual drywell pressure below the high pressure trip setpoint. As a result, post-scram recovery operations were complicated in both events. With the information provided, BWR licensees may consider changes to either the high drywell pressure load shedding logic for the drywell coolers (e.g., as at the Quad Cities facility) and/or the high drywell pressure override arrangements.

(5) <u>It is suggested that an industry representative or group, such as INPO, take appropriate actions to identify and feedback to BWR facilities the corrective measures needed to prevent damage to safety/relief valve tailpipe vacuum breakers.</u>

A review of recent operating experience reports reveals that a significant number of safety/relief valve tailpipe vacuum breakers have failed to operate properly at several BWRs. An evaluation of these failures indicates that a potential generic problem may exist with these valves since failures have been reported at four different reactors involving three different plant-sites. Although the failed valves are different in size and manufacturer, the damage appears quite similar from plant-to-plant. It is also significant that a redesign of the valve internals was included in the repair of all of the failed valves. The most serious consequence so far involved the Hatch occurrence on August 25, 1982. In that event the breaker failure significantly added to the complexity, difficulty and seriousness of post scram recovery. In view of the repetitive valve failures, it is recommended that an industry group, such as INPO, review the adequacy of the current design criteria and test program for SRV tailpipe vacuum relief valves.

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*This suggestion first appeared in the preliminary case study report which was issued for "peer review" comments. After the preliminary case study report was issued, but before it was issued final, the Office of Inspection and Enforcement responded by issuing Information Notice 84-35, "BWR Post-Scram Drywell Pressurization." Information Notice 84-35 fully addresses this suggestion. This suggestion appears in the final report simply as a matter of record.

8. REFERENCES

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3. "Summary of Events Surrounding Hatch Unit 2 Scram on 8/25/82," Georgia Power Company.

4. Telecommunications with Georgia Power Company Hatch Unit 2 plant operations personnel, March 3, 1983.

5. Meeting with Georgia Power Company Hatch Unit 2 Plant operations personnel, April 14, 1983.

6. Edwin I. Hatch Nuclear Plant Unit 2, Final Safety Analysis Report, Georgia Power Company, Docket No. 50-366.

7. Georgia Power Company, Docket No. 50-366, Licensee Event Report No. 82-101, September 28, 1982.

8. Georgia Power Company, Docket No. 50-366, Licensee Event Report No. 82-91, September 23, 1982.

9. "Safety Concerns Associated With Pipe Breaks in the BWR Scram System" NUREG-0785, May 1981.

10. "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," NEDO-24342 General Electric Company, April 1981.

11. "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," NUREG-0803, August 1981.

12. "Emergency Procedure Guidelines, Revision 2" NEDO-24934, June 1982.

13. "Emergency Procedure Guidelines, Revision 3" NEDO-24934, December 1982.

14. USNRC IE Information Notice No. 81-28, "Failure of Rockwell-Edward Main Steam Isolation Valves," September 3, 1981.

15. Power Authority of the State of New York, Docket 50-333, Licensee Event Report No. 82-047, October 27, 1982.

16. Power Authority of the State of New York, Docket No. 50-333, License Event Report No. 84-057, January 11, 1983.

17. "Carolina Power and Light Nuclear Power Stations MSIV Stem-Disk and Disk-Piston Disassemblies During Plant Operation" Technical Report, Rockwell International, August 7, 1981.

18. Letter from N. Kalivianakis, Commonwealth Edison Company to J. Keppler, NRC/PIII, transmitting "Quad Cities Nuclear Power Station Units 1 and 2 Special Report, Loss of Auxilliary Electrical Power Event of June 22, 1982." July 9, 1982.

19. Carolina Power and Light Company, Docket No. 50-325, Licensee Event Report No. 79-74, November 16, 1979.

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21. Letter from W. A. Widner, Georgia Power Company to the ONRR, USNRC, February 26, 1981.

22. Letter from J. Stolz, USNRC to J. Beckham, Georgia Power Company, September 1, 1981.

23. Letter from J. Beckham, Georgia Power Company to the Director of ONRR, NRC, October 1, 1981.

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25. Letter from R. Starostecki, USNRC/RI to W. Harrington, Boston Edison Company, transmitting Inspection Report No. 50-293/82-30, dated January 14, 1983.

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27. Letter from R. Staffa, Georgia Power Company to J. O'Reilly, USNRC/RII, March 22, 1970.

28. AEOD Engineering Evaluation, "Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting" (AEOD/E322), September 21, 1983.

29. Georgia Power Company, Docket No. 50-321, Licensee Event Report No. 83-112, December 15, 1983.

30. Edwin I. Hatch Nuclear Plant Unit 1, Final Safety Analysis Report, Georgia Power Company, Docket No. 50-321.

31. Letter from J. Stolz, USNRC to J. Beckham, Georgia Power Company, January 4, 1984.