MAY 17, 1982

LRG–II POSITION PAPERS

VOLUME V

TECHNICAL DISCUSSIONS AND RESOLUTIONS OF 7 LRG–II ISSUES. THE POSITIONS TAKEN IN THESE PAPERS WILL BE REFERENCED IN LRG–II PLANT OL APPLICATIONS.

NOTE: VOLUME V COMPLETES RESPONSES TO THE 56 LRG–II ISSUES. THIS VOLUME ALSO CONTAINS REVISED LRG–II POSITIONS FOR SEVERAL PREVIOUSLY–FILED ISSUES.

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MJA:hmc/D051815
5/18/82

1-RSB

REQUIREMENT FOR AUTOMATIC RESTART OF HPCS AFTER MANUAL TERMINATION

ISSUE:

TMI Action Plan Item II.K.3.21 required analysis to determine if the LPCS and LPCI systems should not be modified to provide automatic restart. The submittal indicated that a relatively straightforward HPCS design modification could be made to automate the restart of HPCS on low vessel level following manual termination of this system. The NRC Staff required a commitment to install the HPCS restart modification.

LRG-II POSITION:

The LRG-II position is to not modify the HPCS logic to automatically restart the HPCS pump on low reactor vessel water level following manual termination of the system. This revised position is based on a letter from J. R. Miller (NRC) to D. L. Holtzscher (LRG-II) dated February 26, 1982 which indicated that this HPCS logic modification is optional. The rationale for not implementing this modification is contained in the BWR Owners' Group (BWROG) report, transmitted in a letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC) dated December 29, 1980 which concluded:

"This review has included a consideration of all aspects of HPCS, LPCS and LPCI system operation which would be influenced by any expanded automatic restart capabilities. It is concluded that the current system design is adequate and no design changes are required. This conclusion is based on a combination of factors that include: the comprehensive nature of BWR operator training, the emphasis placed in this training on reactor water level control, the Emergency Procedure Guidelines, the relatively long time the operator has to correct errors and the extent to which low reactor water level conditions are displayed and alarmed in the control room. The most important consideration is that the benefits of providing enhanced automatic ECCS reinitiation do not justify the associated penalties of increased system complexity, reduced system reliability, restricted operator flexibility and the undesirable effects discussed in this memorandum."

LRG-II POSITION PAPER
May 17, 1982
REVISION 1

8-RSB

ASSURANCE FOR LONG TERM OPERABILITY
OF THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS)

ISSUE:

TMI Action Plan Item II.K.3.28 identified the need to assure that air or nitrogen accumulators for the ADS valves are provided with sufficient capacity to cycle the valves open five times at design pressures. The long-term air supply must also be designed to withstand a hostile environment and still perform its function 100 days after an accident.

Since the time when the ADS would be needed during or after an accident is dependent upon a variety of scenario-specific unknowns such as equipment availability, operator actions, break size, etc., it is unacceptable to NRC to allow the ADS to be unavailable anytime the reactor is pressurized.

Leakage through the accumulator check valves must not disable the ADS before action is taken to provide the back-up air supply. No single active failure may disable the long-term air supply.

LRG-II RESPONSE

In order to demonstrate ADS valve operability, a generic discussion on ADS valve accumulator capability is provided. Individual plant descriptions of backup air supply system functional design, operator actions to establish the backup air supply, and surveillance testing are also provided. In all cases, no single active failure can disable both divisions of the safety grade air supply nor can the loss of the non-safety grade air supply to the accumulator disable the ADS before action can be taken to activate the safety-grade air supply.

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The ADS uses selected safety/relief valves for depressurization of the reactor. Each of the safety/relief valves (S/RV) utilized for automatic depressurization is equipped with an air accumulator, a check valve, and a safety grade backup air supply to preserve pressure. The safety grade ADS pneumatic supply is separate for the two divisions. One supplies the ADS valves on steamlines "A" and "C", the other supplies to the ADS valves on steamlines "B" and "D". The air supply to the ADS valves has been designed such that the failure of any one component will not result in the loss of air supply to more than one nuclear safety-related division of ADS valves. The loss of air supply to one division of ADS valves will not prevent the safe shutdown of the unit. For all BWR/6's, only three of the ADS valves in one division need to function to meet short-term demands and the functional operability of only one ADS valve will fulfill longer term needs.

I. ADS ACCUMULATOR CAPABILITY

The ADS accumulators are designed to provide two S/RV actuations at 70% of drywell design pressure, which is equivalent to five actuations at atmospheric pressure. The ADS valves are designed to operate at 70% of drywell design pressure because that is the maximum pressure for which rapid reactor depressurization through the ADS valves is required. The greater drywell pressures are associated only with the short duration primary system blowdown in the drywell immediately following a large pipe rupture for which ADS operation is not required. For large breaks which result in higher drywell pressure, sufficient reactor depressurization occurs due to the break to preclude the need for ADS. One ADS actuation is sufficient to depressurize the reactor and allow inventory makeup by the low pressure ECC systems. For conservatism, the accumulators are sized to allow two actuations at 70% of drywell design pressure.

In order to demonstrate the ADS accumulator capability to provide two valve actuations under accident conditions, an equivalent test will be conducted during the Startup Test Program. This test will verify the accumulator's capability to provide five actuations under normal drywell pressure. However, the availability of the safety grade backup air supply system will preclude the need to rely on accumulator capability alone.

II. BACKUP AIR SUPPLY SYSTEM FUNCTION DESIGN

Clinton Project

The normal air supply to the accumulators for the ADS valves and non-ADS safety relief valves is from the station instrument air (IA) system. Compressed air for this system is supplied at 120 psig from one of the three 100% capacity service air (SA) system compressors and processed through one of the three 100% capacity IA system filter/dryer packages. Instrument air to the ADS and non-ADS valves is processed through twelve 20% capacity air amplifiers which double the regulated supply pressure of 80 psig to 160 psig and deliver it to the valve accumulators.

If the normal air supply is not available, the safety-grade backup air supply system will preserve ADS valve accumulator pressure. This backup system has two independent air storage facilities located in separate corners of the basement of the auxiliary building. Each facility consists of eight 1.75 ft^3 bottles, pressurized to 2400 psig, and equipped with appropriate regulatory valves and interconnecting piping to supply one division of ADS valves with a seven day supply of air. Both facilities have remote makeup capability to assure a 100 day post-accident ADS air supply.

The bottles at the air storage facilities are manufactured to D.O.T. Specification 3AA and are equipped with Seismic Category I restraints. ADS valve accumulators, interconnecting piping back to the storage facilities and associated valves are designed to the requirements of ASME Section III, Class 3, and are Seismic Category I. The four motor operated valves and controls for bringing the backup air system into service are powered from Class IE power supplies. The valves and controls for each of the independent air storage facilities are powered from a separate electrical division. The backup air supply system from the air storage facility to the ADS valve accumulators will be environmentally qualified in accordance with the requirements contained in NUREG-0588.

In the event of a normal air supply system problem, one or more of the following control room alarms would be activated:

Trouble with IA Dryer (separate alarm for each dryer)

Trouble with SA Compressor (separate alarm for each compressor)

Low Pressure (70 psig) in Ring Header (separate alarm for each of the six ring headers)

Auto Start of Standby SA Compressor (80 psig)

Not Available IA/SA System (separate alarm for Division I and II)

Low ADS Valve Accumulator Pressure (pressure below 140 psig for any accumulator)

These alarms would alert the operator to a problem in the normal air supply system for the ADS valves.

The operator would verify that automatic actions to maintain the normal supply have occurred and manually perform any which have not. If the normal air supply system cannot be maintained and air system pressure drops from 120 psig to below 70 psig (140 psig to ADS accumulators), then the control rod drive system scram valves will fail open, causing the associated control rods to insert and thus shutdown the reactor. The control room operator places the backup air supply into service by closing the two normal air system supply valves and opening the two backup system supply valves.

In the event of an accident or transient which would result in a containment isolation, the normal air system would be isolated and the backup system automatically placed in service.

Surveillance and testing of the compressed air supply systems for the ADS will consist of the following activities:

A. Control room operator continual surveillance of control room alarms for SA compressors, IA dryers, IA ring header pressures, and ADS accumulator pressure;

B. Auxiliary equipment operator daily inspection of backup air storage pressure;

C. Operability testing of all SRV accumulator check valves to assure proper functioning in accordance with the requirements of ASME Section XI, Subsection IWV.

Perry Project

The normal air supply for the ADS valves' accumulators is from an air compressor located in the auxiliary building. This compressor supplies air to two in-line air receiver tanks located in the intermediate building which have a volume of 10.5 ft^3 each. The compressor automatically maintains air pressure in these two ranks between 2250 and 2500 psig. Each tank serves one division of ADS valve accumulators via a 2500/150 psig pressure regulating valve.

If the air compressor is not available, the compressed air in the two in-line receiver tanks serves as the backup air supply and can recharge the ADS valve accumulators to provide makeup for any system leakage for a period of seven days. Both tanks have a connection on downstream piping to permit commercially available air or nitrogen bottles to be connected to the system to assure a 100 day post-accident ADS air supply.

The ADS air supply system, from the two in-line check valves located upstream of each air receiver tank to the valve accumulators, is designed to the requirements of ASME Section III, Class 3 and are Seismic Category I. The section of this line penetrating the containment and the inboard and outboard isolation valves are designed to the requirements of ASME Section III Class 2 and are Seismic Category I. Since the backup air supply is in-line, the use of motor-operated valves to bring the receiver tank supply into service is not required. The components of the backup air supply system will be environmentally qualified in accordance with the requirements contained in NUREG-0588.

In the event of a loss of air supply from the air compressor, one or more of the following control room alarms would be activated:

Receiver tank air pressure low (2000 psig).

Air compressor/purifier package inoperable.

When the alarm in the control room indicates low receiver tank pressure, the air compressor is manually started and runs until the system pressure is returned to the normal operating range. When the alarm indicates the compressor is inoperable, receiver tank pressure is monitored while the compressor problem is evaluated. If the compressor cannot be restarted in a timely fashion, then commercially available air or nitrogen bottles can be connected to the safety class connections near the air receiver tank to supplement the tank's supply during repairs.

Surveillance and testing of the air supply system for ADS will consist of the following activities:

A. Control room operator surveillance of control room alarms and pressure indication to assure adequate ADS header air pressure.

B. Scheduled checks made to assure that receiver tanks' pressure integrity is maintained and that pressure regulating valves operate.

C. Periodic testing to assure that, upon loss of the normal air supply, ADS header pressure will not decrease at a rate which would jeopardize the capability of the system to maintain ADS header pressure for long-term post accident conditions (i.e., leak rate from receiver tank system will allow commercial air bottles to be added to the system in time to maintain adequate air pressure).

River Bend

The normal air supply for the accumulators of the ADS valves is from the Penetration Valve Leakage Control System (PVLCS). This safety grade system has two independent equipment trains located in the auxiliary building. Each train consists of a filter, compressor, after-cooler, moisture separator and accumulator tank which supply air at 150 psig to one division of ADS valves. The system is designed to remain operational for 30 days following a LOCA.

The PVLCS is designed to seismic Category 1 requirements and is Safety Class 2. Each train is independently powered from an onsite divisional power source. The system will be environmentally qualified in accordance to the requirements contained in NUREG-0588.

Surveillance and testing of the compressed air supply systems for the ADS will consist of inspections and operability testing of the PVLCS.

Air from the PVLCS is dried and filtered by ADS air dryers prior to entering the ADS accumulators. In the event of excessive pressure drop across the air dryer, a control room alarm will be activated.

Pressure transmitters monitoring the PVLCS accumulators automatically start the PVLCS compressors as needed to makeup system demand and leakage. The pressure transmitter in each ADS supply header is alarmed in the control room to signal a loss of air supply.

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May 17, 1982

9 - RSB

LONG TERM OPERABILITY OF DEEP DRAFT PUMPS

ISSUE:

I&E Bulletin 79-15, dated July 1979, identified problems associated with deep draft pumps found in operating facilities. These vertical turbine pumps are usually 30 to 60 feet in length with impellers located in casing bowls at the lowest elevation of the pump and the motor (driver) located at the highest elevation, with the discharge just below the motor. This configuration has experienced excessive vibration and bearing wear which has been attributed to:

- o Flexibility of the rotor and casing structure.
- o Natural vibration frequencies occuring near the operating speed of the pump.
- o Flow inlet conditions conducive to the formation of vortices at the bellmouth of the pump.
- o Misalignment between the shaft and column.

These conditions can cause and aggravate vibration induced wear of the pump components suggesting that these pumps might not be able to perform their required functions during or following an accident.

LRG-II plants must define a program and provide data which confirm long term operability of deep draft pumps including the ECCS pumps (HPCS, LPCS and RHR). An acceptable program would meet the NRC document "Guidelines for Demonstration of Operability of Deep Draft Pumps" which was transmitted to LRG-II plants. These guidelines include:

- o Following good installation procedures including optical alignment and proper torquing sequences.
- o Installation of extensive flow, pressure and vibration instrumentation.
- o Three phases of testing, disassembly, measurement and evaluation to determine the pump's acceptability.

LRG-II POSITION:

Inherent design features of deep draft pumps at LRG-II plants preclude the undesirable conditions described above. Long term operability of LRG-II deep draft pumps will be maintained by:

1. Frequent functional and vibration surveillance testing.

2. Performing preventive maintenance and inspection on a scheduled frequency.

3. Analyzing and evaluating functional data for off-normal trends.

BACKGROUND:

Conditions that cause and aggravate vibration induced wear do not exist in LRG II plant deep draft pumps.

ECCS Pump Design

The LRG-II plants use Byron Jackson (BJ) pumps in ECCS service as listed in Table 9-RSB. Each Byron Jackson ECCS pump is supplied with a casing or suction barrel. These pumps are not installed in a wet sump from which they take suction as in the case with the vertical turbine pumps described above.

These pumps do not use long limber columns typical of some other deep draft pumps. The LRG-II BJ pumps are significantly shorter than the 30 to 60 foot pumps described in IEB-79-15 and thus demonstrate significantly fewer of the problems associated with longer pumps. The longest BJ ECCS pump in service in an LRG-II plant is 26 feet long. The rigidity of the pump assembly is enhanced by the use of seismic rings between pump assembly and the barrel.

The hydraulic design has been developed over the last forty years of experience in many applications. The pumps use a double suction first stage to provide stability over a wide range of flows. Column frequencies are well removed from the pump speed.

The barrels are relatively large in diameter thereby providing low velocities around the pump inlets. The suction barrels include seismic restraints which are of pin or spoke configuration which act as flow straighteners to supress vortex formation. All pumps are provided with high precision, keyed sleeve-type couplings.

Safety Grade Service Water Pump Design

The LRG-II Plants use safety grade service water pumps made by various manufacturers as listed in Table 9-RSB. The rigidity of the pump assembly is enhanced by the use of seismic supports between the pump column and nearby support structures.

The hydraulic design has been developed over years of experience in many applications. Column frequencies are well removed from the pump speed.

Vortex breakers in the structure are used to suppress vortex formation. All pumps are provided with high precision, keyed sleeve-type couplings.

Comments on the Pump Operability Guidelines

Based on the previously discussed differences in characteristics between the LRG-II pumps and those described in the NRC guidelines, it is the LRG-II position that the NRC guidelines for assuring long term operability of deep draft pumps are not applicable to LRG-II plants. Furthermore, for the reasons which follow, the NRC guidelines would not result in improved pump operability.

Dimensional checks recommended by the NRC are part of the pre-shipment inspection and need not be repeated in the field. Compliance with bolt torquing procedures is good practice and is followed at LRG-II plants per the manufacturer's recommendation. The use of optical alignment equipment is not necessary. Dial indicators are used to align the motor to the pump. Adequate alignment and shaft straightness during installation and prior to operation are best demonstrated by the ability to turn the assembled pump and motor by hand. Continued alignment or wear of the machine can be determined from periodic vibration test data.

Experience has shown that vibration monitoring can be used to detect pump degradation. The use of vibration monitoring instrumentation is recommended by the manufacturer and Hydraulic Institute Standards. Such instrumentation will detect significant pump problems and will, if monitored during periodic surveillance testing in conjunction with hydraulic performance, provide a total assessment of pump conditions. If the readings are initially acceptable and do not show continuous change with operation, the pump is not deteriorating. Vibration monitoring data will be periodically obtained and reviewed at LRG-II plants, as part of the Inservice Inspection (ISI) Program. Installation of instrumentation inside the suction barrel is not condoned by the vendor. This instrumentation could be unrealiable and could damage a pump if it were to come loose.

The measurement of pressure pulses also is not recommended, again because of the concern for the potential of internal instrumentation damaging a pump.

The requirement for three step tests with measurement of bearing clearances, is not considered a viable method of improving pump reliability. A deep draft pump must be completely disassembled to perform such an inspection. This enhances the possibility of misalignment and damage, and has the net effect of starting a new pump at each test step. Furthermore, no wear will be measured unless the pump is not in satisfactory condition; this would have been obvious from the vibration monitoring discussed above, and/or the inability to turn the pump by hand.

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LRG-II PROGRAM FOR ASSURING LONG TERM OPERABILITY OF DEEP DRAFT PUMPS

The basic activities involved in determining operability and providing control and some assurance of long term operability of deep draft pumps are described below.

Preventive Maintenance Program

Preventive maintenance and surveillance testing are scheduled at frequent intervals. Scheduled preventive maintenance consists of obtaining megger (resistance) readings of the motor windings, lubricating critical rotating components, plus general cleaning and inspection of rotating electrical equipment at intervals of 3 months to 18 months. Inspection, overhaul, alignment, and impeller lift adjustments will be scheduled as ISI program test results dictate.

Functional Testing and Surveillance

Each deep draft pump is scheduled to L tionally tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code which currently requires testing at least once each 31 calendar days. Pump inlet pressure, differential pressure, flow rate, vibration, and upper pump bearing temperature measurements will be taken. Engineering analyses are performed to identify changes or pump performance trends that may be indicative of off-normal operating conditions. Functional testing and surveillance requirements are specified in LRG-II Plants Technical Specifications, Surveillance Procedures, and Inservice Inspection Programs.

Vibration Monitoring Program

As part of the LRG-II plant ISI programs, vibration measurements will be taken in accordance with Section XI of the ASME Boiler and Pressure Vessel Code which currently requires monitoring every 31 days. In addition, vibration data bases will be established during the preop/startup testing phase and will be used for comparison purposes during later surveillance testing. Journal bearing wear and shaft whip can be deduced from vibration increases. The data will be evaluated on a scheduled basis to predict potential bearing and journal failures and establish replacement schedules. Data will be available on site for inspection.

TABLE 9-RSB

LRG-II DEEP DRAFT PUMPS CHARACTERISTICS

CLINTON

PLANT APPLICATION	MANUFACTURER	MODEL	CAPACITY	LENGTH	PUMP DIAMETER	SUCTION SIZE
High Pressure Core Spray	Byron Jackson	28 DX-18CKXLH	5010 gpm	26'	60"	24"
Low Pressure Core Spray	Byron Jackson	28 DX21CKXL	5010 gpm	20'11"	62"	20"
Residual Heat Removal (3)	Byron Jackson	28 DX18.5CKXL-2	5050 gpm	20'5"	56"	20"
Shutdown Service Water (2)	Byron Jackson	37 KXL	16,500 gpm	41'8"	42"	NA
Shutdown Service Water	Bingham Pump Co.	8X14A-VCM	1100 gpm	30'4¼"	15"	NA

PERRY

PLANT APPLICATION	MANUFACTURER	MODEL	CAPACITY	LENGTH	PUMP DIAMETER	SUCTION SIZE
High Pressure Core Spray	Byron Jackson	30 DX-19CKX-L12	6250 gpm	23'7"	72"	24"
Low Pressure Core Spray	Byron Jackson	30 DX-20CKX-H4	6250 gpm	22'11"	64"	24"
Residual Heat Removal (3)	Byron Jackson	30 DX-20-CKX-H	7260 gpm	20'11"	60"	24"
Emergency Service Water (2)	Goulds Pump Co.	VIT-20X30BLC	11,500 gpm	41'		33"
Emergency Service Water	Goulds Pump Co.	VIT-8X12JMC	960 gpm	35'6"		$11^3/_8$

RIVER BEND

PLANT APPLICATION	MANUFACTURER	MODEL	CAPACITY	LENGTH	PUMP DIAMETER	SUCTION SIZE
High Pressure Core Spray	Byron Jackson	28 DX18CKXL-15	5125 gpm	16'	60"	24"
Low Pressure Core Spray	Byron Jackson	30 DX21CKXL-4	5125 gpm	25'5"	62"	
Residual Heat Removal (3)	Byron Jackson	28 DX18.5CKXL-2	5165 gpm	20'5"	56"	20"
Standby Service Water Pump (4)	Hayward-Tyler Pump Co.	18X23VSN	7690 gpm	58'9"	26.5"	NA

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LRG-II POSITION PAPER
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Revision 1

3-CPB

CHANNEL BOX DEFLECTION

ISSUE

General Electric report NEDO-21354 describes a channel creep deflection phenomena that may interface with control rod insertion. Long term channel deflection occurs when fuel channels are radiated to high exposures or are located in a region of the core which has a gradient in fast neutron flux. The resulting bulge (caused by long term creep) or bow (caused by differential deflection of the channels) reduces the size of the gap available for control rod insertion.

A program to detect the onset interference between the channel box and the control blade is required. NEDO-21354 describes a control rod drive setting friction test which can be used to measure the interference of the channel with the control blades. This testing should be included in the program or an alternative proposed.

LRG-II POSITION

The LRG-II position is to adopt the following guidelines:

Channel Box Deflection Guidelines

The following general guidelines minimize the potential for and detect the onset of channel bowing:

A. Records will be kept of channel location and exposure for each operating cycle.

MJA:hmc/D051013-1
5/18/82

B. Channels shall not reside in the outer row of the core for more than two operating cycles.

C. Channels that reside in the periphery (outer row) for more than one cycle shall be situated in a core location each successive peripheral cycle which rotates the channel so that a different side faces the core edge.

D. At the beginning of each fuel cycle, the combined outer row residence time for any two channels in any control rod cell shall not exceed four peripheral cycles.

After core alterations (i.e., reload) and before reaching 40% thermal power, a control rod drive friction test* shall be performed for those cells exceeding the above general guidelines or containing fuel channels with exposures greater than 30,000 MWd/T (associated fuel bundle exposures). After the technical specification scram speed surveillance test on each rod, as required by BWR/6 Standard Technical Specification 4.1.3.2.a, each control rod meeting the above conditions will be allowed to settle a total of two notches, one notch at a time, from the fully inserted position.

Total control rod drive friction is acceptable if the rod settles, under its own weight, to the next notch within approximately ten seconds. If the rod settles too slowly, a rod block alarm will actuate, indicating possible impending channel box-control blade interference. The results of this test will be considered acceptable if no rod block alarm is received. This testing will give an early indication of this interference and will prompt an investigation into the source of the friction. If necessary, corrective action will be completed before startup after the next core alteration.

In lieu of friction testing, fuel channel deflection measurements may be used to identify the amount of remaining channel lifetime for channels exceeding 30,000 MWd/T (associated fuel bundle exposures).

In the future, analytic channel lifetime prediction methods, benchmarked by periodic deflection measurements of a sample of the highest duty fuel channels, could be used to ensure clearance between control blades and fuel channels without additional testing.

* This control rod settling friction test, also recommended by GE, provides an equivalent level of the tests described in NEDO-21354. This test provides adequate assurance of the scram function. The amount of friction detectable by this test is ~250 lbs. Control Rod Drive Tests indicate that the CRD will tolerate a relatively large increase in driveline friction (350 lb) while still remaining within technical specification limits. The control blade is in its most constrained, highest friction location when it is fully inserted. The ability of the blade to settle from this position demonstrates that the total drive line friction is less than the weight of the blade (~250 lbs).

MJA:hmc/D051013-3
5/18/82

6-CPB

INSTRUMENTATION TO DETECT INADEQUATE CORE COOLING

ISSUE:

In response to TMI Action Plan Item II.F.2, instrumentation to detect inadequate core cooling should be provided.

LRG II POSITION:

The LRG-II position is to support the BWR Owners Group effort to address detection of inadequate core cooling. The BWROG has undertaken a study of inadequate-core-cooling instrumentation. The study, expected to be completed in July 1982, will assess the feasibility and necessity of providing reliable, responsive instrumentation.

MJA:hmc/D05108
5/18/82

1-ICSB

FAILURES IN VESSEL LEVEL SENSING LINES COMMON TO CONTROL AND PROTECTIVE SYSTEMS

ISSUE:

Operating reactor experience indicates that a number of failures have occurred in BWR reactor vessel level reference sensing lines and that, in most cases, the failures have resulted in erroneously high reactor vessel level indication. For BWRs, common reference sensing lines are used for feedwater control and as the basis for establishing vessel level channel trips for one or more of the protective functions (reactor scram, MSIV closure, RCIC, LPCI, ADS, or HPCS initiation). Failures in such sensing lines may cause reduction in feedwater flow and consequential delay in trip within the related protective channel.

If an additional failure, perhaps of electrical nature, is assumed in a protective channel not dependent on the failed sensing line, protective action may not occur or may be delayed long enough to result in unacceptable consequences. This depends on the logic for combining channel trips to achieve protective actions.

It is our position that those reference lines common to the feedwater control function and to any of the protective functions for loss of feedwater events be identified and that the consequences of failures in such reference lines concurrent with the worst additional single failure in the protective systems (reactor scram, MSIV closure, ADS, RCIC, HPCS/HPCI, LPCI, etc.) or their initiation circuits by analyzed.

MJA:rf:rm/150B30

LRG II RESPONSE:

Relay Plants (Perry, River Bend)

The following assessment of a break in a vessel level sensing line, common to control and protective systems, in combination with the worst single failure in a protective channel shows the resulting accident is less severe than, and bounded by, the accidents described in Chapter 15 of the Perry and River Bend FSARs. This conclusion is based on a detailed analysis of a 251 size BWR/6. A comparison of the characteristic of the 218 and 238 size plants shows the analysis for the 251 size plant to be conservative.

All combinations of vessel level instrument line breaks and active single failures in the remaining three electrical divisions were examined to determine which combination had the most severe consequences. The postulated failure path with the most severe consequences is: failure of the Division 1 instrument line, from which feedwater is controlled, combined with an electrical failure in Division 3 Reactor Protection System (RPS) scram circuits. This combination will prevent the plant from scramming on Level 3.

The sequence of events is shown in Table 1-ICSB-1. The high Division 1 level indication input into the feedwater controller causes feedwater to coast down to zero flow. Level drops because the reactor is steaming at nearly full power but has no makeup water. When water level reaches Level 3 a scram would normally occur but, because of the assumed failures, it does not. Scram does not occur because complimentary failures in the "one out of two, twice" logic, (Division 1 or Division 3) and (Division 2 or Division 4), are assumed. The line in Division 1 is assumed to break in such a way that neither Level 3 nor Level 8 scram circuitry is initiated. Division 3 RPS circuitry is also assumed to fail. Therefore, it is not possible to activate the "one out of two" logic on the Divisions 1 and 3 side.

Water level will decrease through Level 2 where RCIC and HPCS will initiate and the recirculation pumps will trip. If the reactor does not scram from high drywell pressure or operator action, the water level will continue to drop to a minimum level somewhere above Level 1, still well above the top of the active fuel. The reactor will settle out at an equilibrium power level of about 15% rated power. The turbine will continue running and HPCS and RCIC will provide reactor makeup water.

The core will remain covered at all times and the MCPR will remain above 1.06. No fuel will fail. Automatic scram functions are still available; if the level were to drop below Level 1, the vessel would isolate causing a scram on MSIV position. Low pressure ECCS is always available but is not needed.

The preceding analysis, although done for a 251 size BWR/6, is applicable to the 218 and 238 size plants. The minimum level that the water inventory would reach depends on the following factors:

(1) initial power level and power decay characteristics,

(2) combined HPCS and RCIC flow capacities, and

(3) the bulk water volume above Level 1.

The power decay characteristics are similar for the three plant sizes. The combined HPCS and RCIC flow capacities, as a proportion of rated feedwater flow, are similar for the three plant sizes. However, the bulk water to power ratios for 238 and 218 plants are approximately 3% larger than that for a 251 plant, i.e., relatively more water inventory is available for 238 and 218 plants. This assures that the minimum water level for 218 and 238 plants would not be lower than that for a 251 plant.

MJA:rf:rm/150B32

Solid State Plants (Clinton, GESSAR)

The RPS logic in BWR/6 solid state plants requires an 2-out-of-4 channels to scram. Therefore, if one RPS channel reads erroneously high due to the instrument line failure and any additional RPS channel is assumed to fail, there are still 2 remaining channels left to accomplish normal scram.

Therefore, there will always be a normal Level 3 scram prior to automatic initiation of either (or both) high-pressure coolant injection systems. It is possible to fail RCIC or HPCS by postulating the additional failure in ECCS busses 2 or 3 respectively. However, both systems cannot fail due to a single electrical failure. The postulated worst case scenerio is a break in the reference line on the division that is controlling feedwater in conjunction with a failure of the HPCS. Normally, the operator would switch feedwater control from the failed instrument line to the operable one as soon as the level mismatch is detected by the annunciator alarm. This would immediately restore normal water level.

Should the operator fail to do this, the water level would continue to drop slowly until it reached Level 2. A trip at this level would normally initiate both HPCS and RCIC and trip the recirc pumps. However, assuming the additional electrical failure of HPCS, only RCIC will start. Since a successful scram occurred at Level 3, RCIC is sufficient to cause water level to turn around between Level 2 and Level 1 and rise, slowly refilling the vessel as power decays.

If still unattended, the vessel level will gradually increase until it reaches Level 8. At Level 8, the RCIC turbine will trip and the main turbine stop valves will close. The water level will drop back toward Level 2 and the cycle will repeat itself driven by the ever-decreasing residual heat decay in the vessel. This will limit vessel water level between Level 2 and Level 8 until the operator takes the remaining shutdown action. The postulated scenario therefore has no adverse safety consequences for BWR/6 solid-state plants.

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TABLE 1-ICSB-1

SEQUENCE OF EVENTS

Time (sec)	Events
0	One of the water level reference legs break (assume feedwater control relies on this instrument line). Feedwater starts to decrease due to a false high water level reading in the failed instrument line.
5.0	Feedwater flow decrease to zero.
6.9	Actual level drops to L-3. No low level scram and high power source pump trip due to the failure of the reference leg and an RPS channel.
11.9	Water level drops to L-2, trips the recirculation pumps and also initiates RCIC and HPCS.
33.	HPCS and RCIC flow starts to enter vessel.
71.	Water level reaches minimum and begins to rise. The minimum level is above the L-1 setpoint.
~200	A new equilibrium state is established at ~15% NBR power.

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LRG II POSITION PAPER
May 17, 1982

2-ICSB

REDUNDANCY AND DIVERSITY OF HIGH/LOW PRESSURE SYSTEM INTERLOCKS

ISSUE:

During normal and emergency conditions, it is necessary to keep low pressure systems, that are connected to the high pressure reactor coolant system, properly isolated from the high reactor coolant pressure. Overpressurization of low pressure ECCS lines increases the potential for loss of integrity of the low pressure system which could result in radioactive releases. The two major areas of concern are:

Redundancy of Interlocks on Low Pressure ECCS
Redundant overpressure protection of the low pressure ECCS lines must be provided. The requirements of SRP 6.3 can be met by providng an interlock on the motor-operated ECCS injection valve that prevents the valve from opening until reactor coolant system pressure is below the ECCS design pressure.

Diversity of Interlocks
It is the NRC's position that for low pressure/high pressure system interfaces where two motor operated valves constitute the low pressure/high pressure interface, the valves should have independent and diverse interlocks to prevent both valves from being opened unless the primary system pressure is below the subsystem design pressure.

LRG II POSITION:

Redundancy of Interlocks on Low Pressure ECCS
The solid state (Clinton, GESSAR) design has interlocks that use a pressure reading from the reactor vessel. No modifications are required to meet the requirements of SRP 6.3.

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The present relay design (Perry, River Bend) of the high pressure/low pressure interface of the low pressure ECCS lines is illustrated in Figure 2-ICSB. The pressure interlocks, a pressure transmitter/trip unit between the testable check valve and the motorized injection valve (MOV), is designed to be functionally tested during test opening of the MOV. Automatic initiation of the low pressure ECCS by the LOCA signal will bypass the interlock and immediately open the MOV.

As a result of recent NRC concerns with regard to protecting low pressure piping upstream of the MOV, the LRG II position is to modify the existing relay interlock circuitry. The modification will remove the LOCA signal bypass. Thus the MOV will be interlocked shut for all reactor pressures greater than 450 psi, given a postulated failure of the check valve.

The above modification will change the results of the Appendix K core cooling analysis. Plant unique calculations for Perry have shown that the worst case peak clad temperature (PCT) would increase by 38°F. A similar increase is expected for River Bend. The current PCT margins are sufficient to accommodate this 38°F rise and still be well below the PCT limit of 2200°F.

Diversity of Interlocks (Relay and Solid State)
RHR suction lines incorporate two motor-operated valves as the low pressure/high pressure interface. LRG II plants incorporate a sufficient level of protection to prevent inadvertent opening of these low/high pressure interface valves. Redundant pressure interlocks with continuous on-line monitoring capability and the requirement for frequent monitoring of the trip channel status assures the reliable operation of the automatic protection feature.

LRG II Plants include interlocks which prevent the operator from opening these valves when reactor pressure is high. The trip unit setpoints are set at 135 psig as compared to a pressure rating of 500 psig for the piping. The two isolation valves in the suction line have divisionally separated controls. These valves are manually controlled pressure-interlocked valves. Each valve control circuit has two pressure interlocks either of which will prevent the valve from being open. It would require a failure of all 4 transmitter trip unit channels to permit operation of both valves when the reactor pressure is high.

The interlocks are controlled by analog pressure transmitters which measure reactor coolant pressure and transmit a signal proportional to the pressure to a solid state trip unit and a visual indicator. This design permits on-line monitoring of the transmitter outputs on analog indicators in the control room so that cross comparison of the output values can be made between channels and other control room pressure indicators. Technical Specifications require a channel check of these systems to be made each 12 hours. The trip units are located in the control room for ease of calibration and testing.

In addition to these automatic protection features, administrative controls do not permit placing the RHR system in the shutdown cooling mode until the reactor pressure has been reduced to less than 135 psig. The pressure indications used for determining reactor pressure when placing the system in shutdown cooling are located on the main control panel and are different from those used in the overpressure protection trip system.

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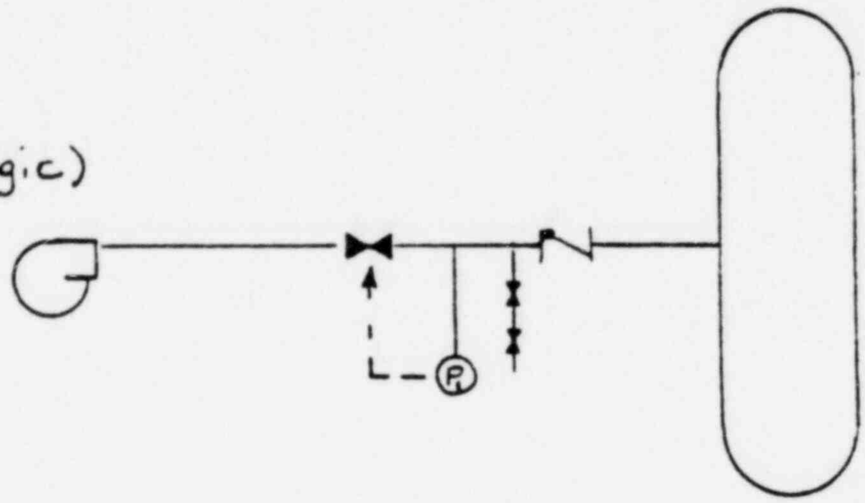
PROPOSED LOW PRESSURE ECCS

LOW/HIGH PRESSURE INTERFACE INTERLOCK

BWR-6

(relay logic)



OPEN PERMISSIVE ON INJECTION VALVE

IF P_I < 450 psig

FIGURE 2-ICSB

2-CSB
HYDROGEN CONTROL CAPABILITY

ISSUE:

Provide a description of the program to improve the hydrogen control capability.

The program should include:

1. a description of the system the plants propose to install,

2. the installation schedule,

3. its design bases and,

4. research programs (including schedules) designed to demonstrate and/or confirm efficacy of the proposed system.

LRG-II POSITION:

The LRG-II position is to participate in the Hydrogen Control Owners Group (HCOG program) to improve hydrogen control capabilities for Mark III containments. This position does not apply to GESSAR.

A hydrogen control program document was submitted on behalf of the HCOG in a letter from J. D. Richardson, HCOG Chairman to H. R. Denton dated January 15, 1982. This report identifies tasks needed to satisfactorily address the use of an igniter system in the Mark III containment. Both generic and plant specific tasks are included. A brief status of each of the tasks identified with respect to LRG-II plants is attached.

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HYDROGEN CONTROL CAPABILITY

A distributed igniter system is proposed for hydrogen mitigation at LRG-II plants. The design for this hydrogen control system would be based on the igniter system previously developed at Grand Gulf, Sequoyah, McGuire and D. C. Cook Nuclear Stations. The system will be installed and operational prior to startup. Because proposed system design details and specific locations are unique for each plant, individual analyses will be documented in the individual LRG-II plant license applications. No further LRG-II group submittals are anticipated.

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STATUS REPORT ON HCOG PROGRAM TASKS FOR LRG-II PLANTS

Select Scenario - Scenario selection and development of hydrogen release rates have been completed by the HCOG and GE. The GE report evaluating the most probable accident scenario and providing generic hydrogen release rates is scheduled to be submitted in second quarter 1982. This report is applicable to LRG-II plants.

Select Mitigation System - Generic selection criteria were developed by HCOG, based on initial studies by Mississippi Power & Light for Grand Gulf Nuclear Station; the igniter mitigation system was selected. Plants specific design features are being evaluated for the use of igniters at LRG-II plants.

Design Hydrogen Ignition System - Specific igniter designs and generic design criteria for the hydrogen control system were established through HCOG. LRG-II participants will provide plant unique design reports to provide details on the igniters, proposed locations and system operation.

Containment Ultimate Capacity Analysis - LRG-II participants are evaluating the ultimate structural capacity of their containments and plant specific reports will be submitted. Completed analyses have demonstrated capability of Mark III containments to withstand internal pressures in excess of 45 psig.

Selection of Containment Response Analyses Code - LRG-II participants as members of HCOG have completed this task with the development and selection of CLASIX-3 computer program. A report entitled "CLASIX-3 Containment Response Sensitivity Analysis" was submitted by J. D. Richardson on behalf of HCOG, in a letter to H. R. Denton dated January 15, 1982. The

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analysis provided a sensitivity studies of the temperature and pressure response of a Mark III containment to hydrogen burns resulting from operation of an igniter system. This report is applicable to LRG-II plants.

Containment Response Analysis - The HCOG has completed the generic Mark III CLASIX-3 analysis. The sensitivity studies were submitted as described above. Plants specific comparisons to the base case analysis are being performed and will be documented in individual LRG-II reports.

Hydrogen Combustion Testing and Analysis - The HCOG has monitored the industry research and analysis regarding hydrogen, including the results of ice condenser plant effort. Specific analysis and testing to address the hydrogen combustion in the Mark III containment is being considered by HCOG in conjunction with the EPRI testing program. LRG-II plants plan to participate in HCOG generic research effort.

Equipment Survivability Analysis - A generic list of essential equipment which must survive the hydrogen burn is being developed by the HCOG and GE. Analysis will be provided to assure that the equipment will be capable of surviving postulated hydrogen burns at LRG-II plants. If applicable, generic heat transfer models of equipment will be developed by HCOG and used in the evaluation.

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LRG-II POSITION PAPER
May 17, 1982

2-ASB

SAFE SHUTDOWN FOR FIRES AND REMOTE SHUTDOWN SYSTEM

ISSUE:

Identify and submit information on any areas where a generic submittal, to demonstrate compliance with Sections III.G and III.L of Appendix R, would be appropriate.

The only issue for which LRG II has chosen to provide a generic submittal is the compliance of the Remote Shutdown System (RSS) with 10CFR50 Appendix R Section III.L, Paragraph 2.d. This paragraph states: "The process monitoring function shall be capable of providing direct reading of the process variables necessary to perform and control the above functions". The above function of concern is stated in Paragraph 2a: "The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions". The NRC staff would like to have all reactors to include the Source Range Monitor (SRM) indication on the RSS panel to satisfy the requirement for monitoring.

LRG II POSITION:

The current LRG II plant designs fully meet the requirements of 10CFR50 Appendix R, Section III.L, Paragraph 2.d. No reactivity indications are needed, to monitor control of reactivity when proceeding to cold shutdown from the remote shutdown system (RSS). Since no parameters are needed, none are displayed on the RSS.

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A scram would be initiated either by operator action or by one of any number of automatic trips before the operator is forced to evacuate the control room. This would assure the control rods are inserted. If the operator is forced to scram the plant and evacuate the control room, power to the scram circuitry will be removed either automatically or by procedure. After the scram, the withdrawal of the control rods by either manual action or by a fire related control system failure would require the scram solenoid breakers, located in the auxiliary building, to be manually reset. Since these solenoids would not be reset during a fire, this would assure that the rods would remain inserted. The negative reactivity inserted on scram, in a BWR, is sufficient to assure the plant can reach cold shutdown with no further reactivity control.

The operator needs to take no action to fully control reactivity (i.e., leave the control rods in) after he has left the control room. Since no action is required, no process variable needs to be displayed.

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2-HFS

EMERGENCY PROCEDURES REACTIVITY CONTROL GUIDELINES

ISSUE:

Develop a generic reactivity control guideline which can be utilized for preparing an emergency operating procedure for an anticipated transient without scram (ATWS) event.

LRG-II POSITION:

It is the LRG-II position to support the development of generic reactivity control guideline for preparing an emergency operating procedure fo. responding to an ATWS event. The reactivity control guideline is currently being developed by these Emergency Procedures Committee of the BWR Owners Group.

The incorporation of the reactivity control guidelines into the next revision of the emergency operating procedure guideline is currently underway. This revision is scheduled for submittal from the BWR Owners Group to the NRC in June 1982. Sponsors of this effort include all LRG-II projects.

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3-CHEB

ESTIMATION OF FUEL DAMAGE FROM POST-ACCIDENT SAMPLES

ISSUE:

A procedure for relating post-accident radionuclide concentrations in reactor coolant and suppression pool samples should be developed.

LRG-II POSITION:

LRG-II projects will prepare plant procedures for estimating fuel damage from post-accident samples. Currently, interim procedures from other projects are under review. In addition, investigation of radionuclide release fractions for the different categories of fuel damage (i.e., cladding failure, overheating, meltdown, etc.) are being conducted with assistance from General Electric Company.

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