CENTRAL FILES

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SHOREHAM NUCLEAR POWER STATION

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January 2, 1980 SNRC-455

Mr. Boyce H. Grier, Director
U. S. Nuclear Regulatory Commission, Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Response to NRC I&E Bulletin 79-08
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Gentlemen:

Enclosed herewith is the Shoreham response to IE Bulletin 79-08. We trust that this submittal, in conjunction with General Electric NEDO-24708, Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, dated August 1979, will be responsive to your request.

Should you require additional information, please do not hesitate to contact us.

Very truly yours,

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

JPM/cc

Enclosure

cc: W/Enclosures

Mr. V. Stello
NRC Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

Mr. J. Higgins (NRC Inspector)

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IE BULLETIN 79-08

ITEM 1

Requirement

Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions parameters and take appropriate corrective action.

b. Operational personnel should be instructed to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 5a of this bulletin); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

Response

A lesson plan has been prepared on the circumstances that led to Three Mile Island (TMI) accident as described in the above mentioned documents. Based on this lesson plan, lectures have been conducted to thoroughly instruct licensed operators, supervisors and managers with operational responsibilities, in the importance of the following items:

a. Basic understanding of the operation of a PWR.
b. Sequence of Events which occurred at TMI.
c. The extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at TMI and other actions taken during the early phases of the incident.

d. The apparent operational errors which led to the eventual core damage.
e. The necessity to systematically analyze plant conditions and parameters and not to make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.
f. Not to override automatic actions of engineered safety features (ESF) unless continued operation of ESF will result in unsafe plant conditions. Refer to Section 5 of this bulletin.

The lesson plan as well as the participation in the formal lectures has been documented in the plant's training files.

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ITEM 2

Requirement

Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate containment isolation, whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

Response

A review of the containment isolation initiation design and procedures has been performed. Containment isolation of lines which do not degrade needed safety features or cooling capability is initiated either automatically or manually upon safety injection system (SIS) initiation signals.

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ITEM 3

Requirement

Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., (RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure, by which this action is taken in a timely sense.

Response

The auxiliary heat removal systems provided to remove decay heat from the reactor core and suppression pool following a loss of the feedwater systems are:

High Pressure Coolant Injection (HPCI) System
Reactor Core Isolation Cooling (RCIC) System
Core Spray System (CS)
Residual Heat Removal (RHR) System
Reactor Water Cleanup (RWCU) System
Control Rod Drive (CRD) System

The description that follows details the operation of the systems needed to achieve initial core cooling followed by suppression pool cooling and then followed by extended core cooling for long term plant shut down.

I. INITIAL CORE COOLING

Following a loss of feedwater and reactor scram, a low reactor water level signal (level 2) will automatically initiate main steam line isolation valve closure. At the same time this signal will put the HPCI and RCIC Systems into the reactor coolant make-up injection mode. These systems will continue to inject water into the vessel until a high water level signal (level 8) automatically trips the systems. Steam is released to the suppression pool through the safety relief valves.

Following a high reactor water level 8 trip, the HPCI System will automatically re-initiate when reactor water level decreases to low water level 2. The RCIC System must be manually reset by the operator in the control room before it will automatically re-initiate after a high water level 8 trip.

The HPCI and RCIC Systems have redundant supplies of water. Normally they take suction from the condensate storage tank (CST). The HPCI System suction will automatically transfer from the CST to the suppression pool if the CST water is depleted or the suppression pool water level increases to a high level.

The RCIC System suction must be manually transferred from the CST to the suppression pool using controls in the main control room. This action would be taken when control room alarms indicate low CST or high suppression pool water level.

The operator can manually initiate the HPCI and RCIC Systems from the control room before the level 2 automatic initiation level is reached. The operator has the option of manual control after automatic initiation and can maintain reactor water level by throttling system flow rates. The operator can verify that these systems are delivering water to the reactor vessel by:

a) Monitoring reactor water level increases when the systems are initiated.

b) Observing system flow using flow indicators in the control room end.

c) Observing system flow is to the reactor by checking control room position indication of motor-operated valves.

Following are the automatic actions which occur in the loss of feedwater event and upon a level 2 (low water) initiation:

A. <u>HPCI</u>

<u>Automatic Actions</u>

a. Steam supply to turbine is automatically lined up.

b. Turbine auxiliaries are automatically started/lined up.

c. Suction flow path is pre-established by the system standby lineup and auto transfers from the condensate storage tank to the suppression pool on low level in condensate storage tank or high level in suppression pool.

d. Injection flow path is automatically lined up.

e. Turbine speed and flow rate are established automatically.

The HPCI System will inject water into the reactor at a flow rate established by the flow controller until manually tripped or automatically tripped by reactor high water level (Level 8). The High Level Auto trip will reset: the HPCI will re-initiate when Reactor level again falls to Level 2.

NOTE: The proper action is for the operator to adjust HPCI flow to maintain reactor level by taking manual control of the flow controller. Should a Level 2 be received while the operator has HPCI in manual control, the HPCI system will automatically revert to full flow taking control away from the operator.

B. RCIC

The automatic operations of RCIC are identical to that of HPCI described above with the following exceptions.

1. The RCIC System suction must be manually transferred from the CST to the suppression pool using controls located in the main control room. This action would be taken when control room alarms indicate low CST or high suppression pool water level. A low CST alarm annuciates when CST level has dropped to 100,000 gal. This allows approximately 3 hours of operation while the operator transfers suction to the suppression pool, a sequence which only takes a few minutes.

2. When RCIC trips on reactor high level (Level 8) it will not automatically reset and re-initiate when level returns to Level 2. The operator must manually:

   a. Close the steam supply valve.

   b. Close the turbine trip/throttle valve to reset the stop valve.

   NOTE: On a loss of feedwater due to Main Steam Isolation or Feedwater Pump trip, the RCIC System should be manually started before either HPCI or RCIC is automatically started. If the RCIC System cannot maintain Reactor level and/or a loss of coolant situation exists, HPCI can be manually initiated. In any event, with no operator action, both systems will automatically initiate at Level 2.

C. RESIDUAL HEAT REMOVAL-STEAM CONDENSING MODE

Manual Action

With RCIC in operation and the feedwater system inoperable, it is desirable to use the RHR System in the steam condensing mode to control reactor pressure. Reactor steam is supplied to RHR heat exchangers from the HPCI steam supply

line. During the initial phases of operation steam is being released to the suppression pool through the safety relief valves. Condensate from the RHR heat exchangers is flushed to the suppression pool while warming up the system and establishing water level in the heat exchangers.

After about 30 minutes, full decay heat steam flow can be accommodated in both heat exchangers, during which time RCIC will be controlled in auto or manual to pump all condensate to the feedwater lines. After about 1.5 hours of operation in this mode, one RHR heat exchanger loop will be able to carry the entire heat load and the other loop may be placed in the suppression pool cooling mode.

Manual actions necessary to achieve this configuration are summarized below (assuming RHR is initially in standby status):

1) Close heat exchanger inlet and outlet isolation valves.

2) Open heat exchanger vents.

3) Set heat exchanger level controllers in manual (75% level) and RCIC pressure controller to 45 psig.

4) Supply service water flow through heat exchanger.

5) Admit steam to heat exchanger using PIC in MANUAL.

6) With LIC, lower heat exchanger level with flow to suppression pool as required to maintain a condensing rate which will keep reactor pressure steady.

7) Adjust heat exchanger level as required to maintain a condensing rate which will keep reactor pressure steady.

8) Shift heat exchanger drain to RCIC when conductivity is in specification.

9) Should LPCI initiation (Level 1) be required while in steam condensing mode, RHR will automatically revert to LPCI mode. No Operator action required.

D. CORE SPRAY SYSTEM

The Core Spray System can be used as an auxiliary heat removal system provided Reactor pressure is below the pump discharge head of the Core Spray pumps. Reactor pressure can be reduced automatically upon ADS operation or manually by opening relief valves to the suppression pool.

Automatic Actions

a. The two Core Spray (CS) Pumps automatically start when reactor Level 1 exists.

b. The CS injection valves automatically open when the differential pressure across the injection valve is less than 450 psid.

c. Injection occurs when reactor pressure decreases below CS pump discharge head.

E. RHR - LOW PRESSURE COOLANT INJECTION (LPCI)

LPCI can be used as an auxiliary heat removal system provided reactor pressure is below the pump discharge head of the LPCI pumps. Reactor pressure can be automatically reduced by actuation of the ADS or manually by opening relief valves to the suppression pool.

Automatic Actions

a. The four RHR pumps will automatically start when reactor Level 1 exists.

b. The LPCI injection valves will automatically open when the reactor pressure is less than 500 psig.

c. Injection occurs when reactor pressure decreases below the RHR pump discharge head.

F. CONTROL ROD DRIVE SYSTEM

The CRD system can provice approximately 60 GPM of condensate to the reactor via the CRD mechanisms to makeup for small inventory losses.

Automatic Actions

The CRD pumps trip on reactor Level 1.

Manual Actions

The CRD pumps can be manually started after the above automatic trip.

G. REACTOR WATER CLEANUP

The RWCU System can be used as an auxiliary heat removal system. This sytem can remove a maximum total flow of approximately 200 GPM from the suction line of the reactor recirc pumps and from the reactor bottom head. The water is routed through a regenerative and non regenerative heat exchanger, through a series of

filters and demineralizers and back to the reactor via the feedwater lines.

Automatic Actions

RWCU system isolation occurs by automatic closure of motor operated valves. The isolation valves protect the RWCU system in the event that a RWCU component should malfunction, and ensures the reactor core from uncovering if the cleanup system piping should rupture. These valves isolate upon a reactor low water Level 2 signal.

Manual Actions

Unless automatic isolation has occurred, the RWCU system can manually operated rom the main control.

II. SUPPRESSION POOL COOLING

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start suppression pool cooling. This mode of operation removes heat resulting from steam discharge to the suppression pool. This would be accomplished by placing the RHR System in the containment (suppression pool) cooling mode i.e., RHR suction from and discharge to the suppression pool.

Manual Actions

Assuming the RHR System is in the steam condensing mode, one loop of the RHR System will be placed in the suppression pool cooling mode as follows:

a) Slowly close steam supply PCV - level increases.

b) Close RCIC supply valve.

c) Open heat exchanger inlet and outlet valves and complete filling and venting.

d) Open key locked suppression pool cooling valve.

e) Start RHR pump.

The operator could verify proper operation of the RHR system suppression pool cooling function from the control room by:

a) Observing RHR and Service Water (SW) system flow using system control room flow indicators.

b) Verify correct RHR and SW system flow paths using control room position indication of motor-operated valves.

Even though the RHR is in the suppression pool cooling mode, core cooling is its primary function. Thus, if a high drywell pressure signal or reactor water Level 1 signal is received at any time during the period when the RHR is in the suppression pool cooling mode, the RHR system will automatically revert to the LPCI injection mode. The Core Spray (CS) system would automatically initiate and both the LPCI and CS systems would inject water into the reactor vessel if reactor pressure is below system discharge pressure.

III. EXTENDED CORE COOLING

When the reactor has been depressurized, the RHR system can be placed in the long term shutdown cooling mode. The operator manually terminates the suppression pool cooling mode of one of the RHR containment cooling loops and places the loop in the shutdown cooling mode as follows:

RHR - Shutdown Cooling Mode

Manual Actions

a. Trip selected loop RHR pumps.

b. Close suppression pool suction valves.

c. Open inboard and outboard pump suction shutdown cooling line valves.

d. Start RHR pump, open injection valve to increase flow.

e. Close heat exchanger bypass valve as required to establish cooldown rate.

In this operating mode, the RHR system can cool the reactor to cold shutdown. Proper operation and flow paths in this mode can be verified by methods similar to those described for the suppression pool cooling mode.

It would also be possible, if necessary, to establish a cooling path by holding a Safety Relief Valve (SRV) open and injecting water from the ECCS pumps into the vessel until it

overflowed through the SRV line back into the suppression pool. This alternative requires the long-term availability of the Safety Relief Valves. The Shoreham SRV air supply system has been modified to provide a minimum of 48 hours of operation assuming loss of instrument air at the time of the event. It also includes a provision for supplementing this air supply from outside of the reactor building.

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ITEM 4

Requirement

Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

Response

Reactor vessel water level is continuously monitored by 25 level instruments which provide automatic initiations to various plant engineered safety features and main control room indicators and alarms for use in assessing the need for any manual actions. These instruments provide vessel level measurement for normal, transient and accident conditions, and indicate from 150 inches below the top of active fuel to 400 inches above vessel instrument zero, as shown in Figure 79-08-4.1.

This span of measurement is covered by five major groups of level instruments as follows:

I. Narrow Range Instruments cover the range of 0 to 60" above instrument zero. This group contains nine sensors providing signals to six local panel level indicators, three indicators and one recorder in the main control room, plus various trip logic functions, annunciators and process computer points. Primary use is to monitor normal water level during power operations.

II. Upset Range Instruments cover the range 0-180" above instrument zero. This group contains one sensor feeding signals to one main control room recorder. Primary use is for following abnormal water level increase during transients. No alarms or trip functions originate from this group.

III. Shutdown Range Instruments cover the range from 0-400" above instrument zero. This group contains one sensor providing signals to one main control room indicator. Primary use is for flooding the reactor vessel. No alarms or trip functions originate from this group.

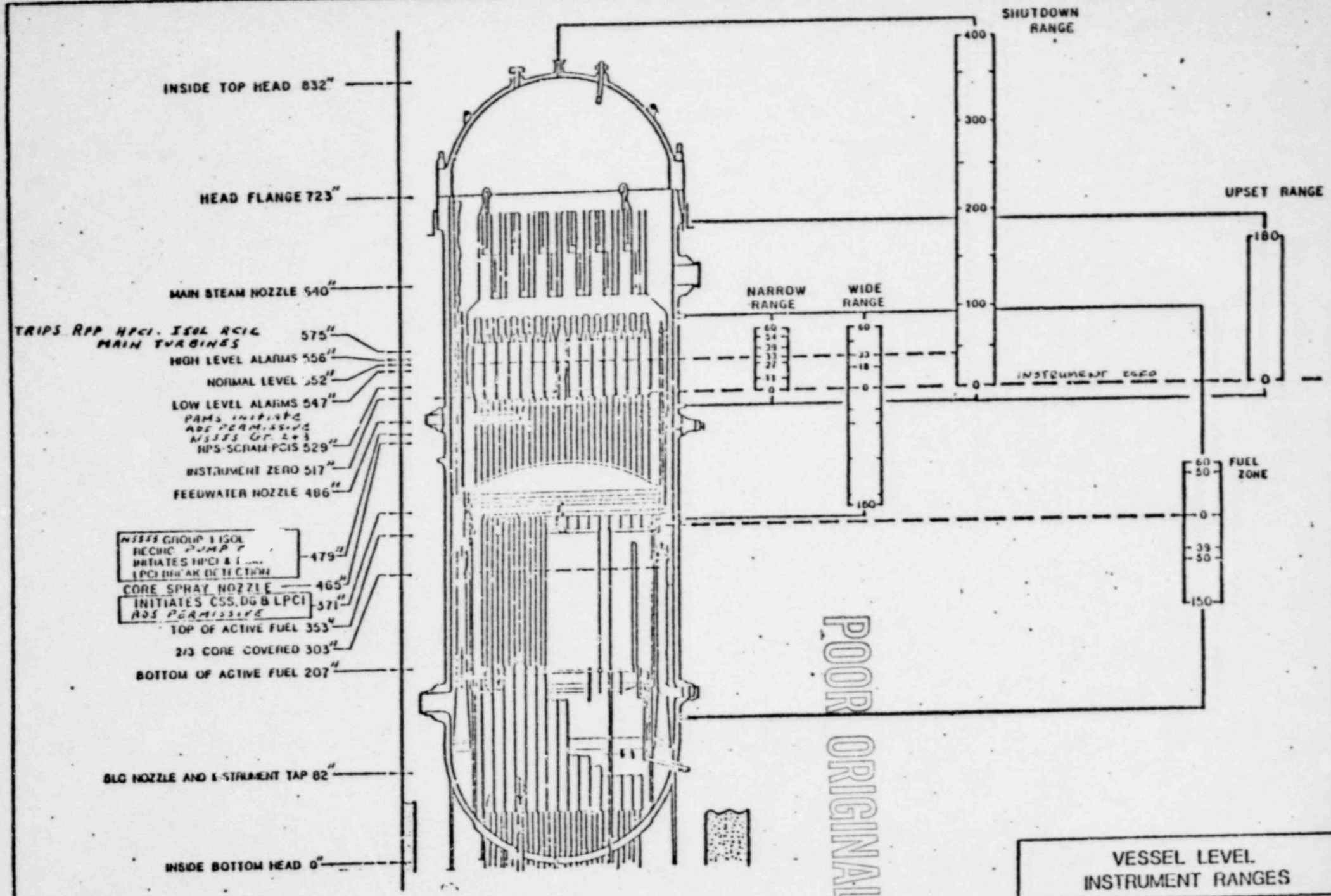
IV. Wide Range Instruments cover the range from -150 to +60" referenced to instrument zero. This group contains 12 sensors providing signals to 12 local panel level indicators, one indicator and 2 post accident monitoring system recorders in the main control room, one remote shutdown panel level indicator, plus various trip logic functions and annunciators.

Primary use is for monitoring water level between the normal operating range and fuel zone.

V. Fuel Zone Range Instruments cover the range from -150 to +50" referenced to an instrument zero corresponding to approximately the top of active fuel. This group contains 2 sensors providing signals to 2 local panel level indicators and one indicator and one recorder in the main control room. Primary use is to provide indication during and after vessel blowdown accident with reactor at 0 psig and the reactor recirculation pumps tripped.

This arrangement of groups of level sensing provides redundancy within a level range, overlap between ranges and function, and allows for confirmation of level conditions from various level sensing sources. This scheme of level measurement devices has operated in BWR plants for 20 years and tests simulated understeam and water line breaks have been conducted showing satisfactory performance.

Instructions to operators to utilize other sources of available information is included in the responses to items 1 and 5 of this bulletin.



VESSEL LEVEL INSTRUMENT RANGES

SHOREHAM NUCLEAR PLANT
LONG ISLAND LIGHTING COMPANY

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IE BULLETIN 79-08

ITEM 5

Requirement

Review the action directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g. vessel integrity).

b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evaluating plant conditions.

Response

A. Operating Procedures, and Training Lesson Plans will be reviewed to ensure that these documents do not contain instructions to override automatic actions of ESF equipment/systems unless continued operation of the ESF equipment/systems result in unsafe plant conditions.

B. A review will be made to ensure operators are provided with instructions not to rely upon vessel level indication alone for manual actions, but to also examine other associated plant parameter indications such as:

Drywell High Pressure or Temperature
Drywell High Radioactivity Levels
Drywell High Humidity
Suppression Pool High Pressure or Temperature
Suppression Pool High Level
Drywell Unit Cooler High Cooling Water Discharge Temperatures
Drywell Unit Cooler High Condensate Flow
Drywell and Equipment Area High Temperature
Drywell Equipment and/or Floor Drain Tank High Fill and Pumpout Rates
Reactor Building Floor Drain and/or Equipment Drain Sump High Fill and Pumpout Rates
Reactor Building Elevation 8' Water Level
Reactor Building Elevation 8' Sump Water Level
Abnormal Reactor Pressure
High Feedwater Flow Rates
Feedwater Flow/Steam Flow Mismatch

High Steam Flow Rates
Safety Relief Valve (SRV) Tailpipe High Discharge Temperature
Safety Relief Valve (SRV) Tailpipe High Discharge Pressure
Reactor Water Cleanup System High Differential Flow
High Process Radiation and/or Area Radiation Levels
Actuation of Various Leak Detection Systems (HPCI, RCIC, Steam Leak Detection).

Other instrumentation that can signal abnormal plant status but not necessarily indicative of loss of coolant are:

High Neutron Flux
Main Turbine Status Instrumentation
Abnormal Reactor Recirculation Flow
High or Low Electrical Current (Amperes) to Pump Motors

The review will be conducted and documented in a check list fashion to ensure that all procedures affected are identified and modified as necessary. This review will be completed 6 months prior to fuel load.

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ITEM 6

Requirements

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Response

A review of safety related valve positions, positioning requirements and positive controls to assure that valves remain positioned in a manner which ensures the proper operation of engineered safety features identified the following four areas:

1. Motor Operated Valves (MOV) - All Category I MOV's have loss of control power relays which provide input to the following: A. Individually to the computer, B. To one of, or a combination of, system inop alarms, system degraded alarms, or loss of control power alarms. Also, if a MOV has a given safety position and it is moved from that position and loses its ability to return automatically, then its respective system inop alarm is sounded.

2. Air Assisted Valves - All air assisted valves are designed to fail in their safety position upon loss of air or loss of electrical power.

3. Modulating Valves - In some Category I process systems it was necessary to provide modulating control during a LOCA. These valves use a 120 v A-C Beck actuator instead of instrument air. There are two failure conditions for these valves: A. Loss of control signal, B. Loss of electrical motive power. On a loss of control signal the valves fail open, close, or as is. On a loss of electrical motive power the valves fail as is. The above failure positions for loss of control signal were determined so as not to cause a loss of system function. If there is a loss of electrical motive power the redundant system provides safety function.

4. Manual Valves - Manual valves are used extensively in Category I systems at Shoreham. Due to their application (test connections, vent and drain lines, instrument root valves, sample connections, and maintenance isolation valves around components such as pumps and heat exchangers), only certain valves have direct remote indication in the main Control Room. However, the operator can determine the positions of the majority of these valves indirectly by use of process instrumentation; e.g., pressure, flow, level, sump levels, etc. In addition, administrative and procedural controls will be used to ensure proper valve position and, in some cases, the valves will be locked in their safety position.

5. A review of the procedures will be conducted as noted in Item 8 of this Bulletin to ensure that valves are returned to their correct safety positions following necessary manipulations.

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ITEM 7

Requirement

Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

Response

All systems designed to transfer potentially radioactive gases and liquids from the primary containment are provided with automatic isolation valves. Upon actuation of containment isolation signals there are no pumps, valves, or equipment which will actuate automatically and allow an inadvertent transfer of radioactive gases or liquids out of the primary containment. In addition, such an occurrence would not be caused by the resetting of the containment isolation signals.

Nevertheless, certain penetrations will be evaluated to determine if high radiation interlocks are required. In addition, the operating procedures for systems designed to transfer potentially radioactive gases and liquids out of the primary containment, such as those listed below, will be reviewed and modified as necessary to assure releases of radioactive liquids and gases will not occur inadvertently.

HVAC - Reactor Building
Drywell Equipment Drainage and Floor Drains
Fuel Pool Cleanup
Reactor Water Cleanup System
Residual Heat Removal
Hydrogen Recombiner

A detail list of primary containment penetrations and associated containment isolation signals is presented in Table 6.2.4.-1 of the FSAR.

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ITEM 8

Requirements

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Response

Plant surveillance testing and maintenance procedures will be reviewed and modified as necessary to require:

a. Verification by test or inspection of the operability of the redundant safety related system prior to removal of any safety-related system from service. Surveillance testing will be performed in accordance with the requirements of the technical specification.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing. The operability of safety-related systems after surveillance testing has been performed will be ensured through adequate instructions and documentation.

c. The explicit notification of involved reactor operational personnel, whenever a safety-related system is removed from and subsequently returned to service, is accomplished through the maintenance request reports and the preventive maintenance scheduled activity worksheet. In addition, any out-of-service safety-related equipment will be procedurally reviewed with the oncoming Watch Engineer at each shift change.

This review will be performed and documented in a checklist manner to ensure that all procedures which render a safety system inoperable are identified and modified as necessary to include the above mentioned requirements. General Operating Procedures are reviewed by the Review and Operations Committee and approved by the Plant Manager to ensure compliance with the requirements of Technical Specifications Limiting Conditions for Operation concerning removal of safety-related systems from service.

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ITEM 9

REQUIREMENT

Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

RESPONSE

The necessary procedures will be prepared in accordance with the requirements of the plant technical specifications and applicable regulatory guides. These procedures will define the conditions under which the reactor would be considered not in a controlled or expected condition of operation. In addition, these procedures will require that upon such a condition an open and continuous communication channel be established and maintained with the NRC.

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ITEM 10

Requirement

Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

Response

Hydrogen gas that may be generated during a transient or other accident may be relieved from the primary system through the Safety Relief Valves (SRV's). There are eleven SRV's located on the main steam lines that relieve to the quenchers located below the suppression pool water level. Since there is a large space between the top of the core and the main steam line nozzles, a significant volume of noncondensable gas can be relieved from the reactor vessel to the suppression pool via that pathway.

The operating modes and procedures for Shoreham's combustible gas control system have been reviewed. It has been found that the system can adequately handle the postulated amounts of hydrogen gas that may be generated during a LOCA. In addition, the combustible gas control system is designed to monitor and control hydrogen released to the containment after a LOCA in accordance with NRC Branch Technical position CSB6-2 and NRC Regulatory Guide 1.7. The system consists of the following four subsystems:

1. Mixing System - The mixing system operates to ensure a well mixed atmosphere in containment and is activated 10 minutes after the LOCA.

2. Hydrogen Sampling System - The sampling system consists of 4 hydrogen analyzers. Two analyzers monitor the drywell and two monitor the suppression chamber and recombiner discharge. The analyzers can be controlled either locally or in the Main Control Room. Redundant drywell Hydrogen Concentration High Alarms are provided in the Control Room.

3. Hydrogen Recombiners System - There are two 150 SCFM thermal type recombiners, each capable of supplying 100 percent capacity. Approximately 9.5 hours after a LOCA the hydrogen

concentration in the drywell would approach the 4 percent limit. Therefore, the Hydrogen recombiners are manually started and are at full capacity within 6 hours after a LOCA. At this time the hydrogen concentration in the drywell will be 3.7 percent. The operator has the controls and instrumentation needed to start up, operate, and shutdown the recombiners in the Main Control Room.

4. Hydrogen Purge System- This system is provided as a backup in accordance with CSB6-2, to the redundant hydrogen recombiners.

An evaluation of the operating procedures for controlling hydrogen generated during accident conditions and for post-LOCA hydrogen recombiner operation will be conducted. This evaluation will be completed 6 months prior to fuel load.

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ITEM 11

Requirement

Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items above.

Response

SNPS Technical Specifications are currently under development with the NRC. No proposed changes have been identified as a result of this initial review.