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THE CINCINNATI GAS & ELECTRIC COMPANY

CINCINNATI, OHIO 45201

August 7, 1979

E. A. BORGMANN
VICE PRESIDENT - ENGINEERING

U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

ATTN: Mr. James G. Keppler, Director

RE: WM. H. ZIMMER NUCLEAR POWER STATION - UNIT 1
NRC IE BULLETIN 79-08
EVENTS RELEVANT TO BOILING WATER POWER REACTORS
IDENTIFIED DURING THREE MILE ISLAND INCIDENT
W.O. 57300, JOB E-5590, FILE # 956, DOCKET # 50-358

Gentlemen:

The attached document is furnished in response to IE Bulletin 79-08. We believe this information provides a complete response to NRC IE Bulletin 79-08.

Very truly yours,
THE CINCINNATI GAS & ELECTRIC COMPANY

E. A. Borgmann

E.A. BORGMANN, SR. VICE PRESIDENT

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RESPONSES TO IE BULLETIN 79-08

ITEM 1. 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; (3) the necessity to systematically analyze plant conditions and 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 conditons (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 TO ITEM 1

All cold license candidates including operators, plant management and supervisors with operational responsibilities have participated in a review of the Three Mile Island Accident. This included the preliminary chronology of the TMI-2, 3/23/79 accident included in I&E Bulletin 79-05A, Enclosure I.

RESPONSE TO ITEM 1a

The circumstances described in I&E Bulletin 79-05, Enclosure I and the understanding of subjects discussed in I&E Bulletin 79-08 Item Ia are being reviewed as follows:

a. A cold license candidate simulator refresher course was conducted in July and August, 1979. The course reinforced and demonstrated BWR level instrumentation design, interpretation, minor transients and upset conditions degrading to loss of coolant conditions. Also covered operator decisions to preclude emergency system component operation.

b. A formal presentation of the events leading to and chronology as now known; with lessons learned be complete by Nov. 1, 1979

RESPONSE TO ITEM 1a CONT'D

c. The continuing onsite training program, Phase II, will provide additional review with operating licensed supervisory and management personnel as further information is made available.

RESPONSE TO ITEM 1b

Station Administrative Directives (SAD's) have been revised to instruct operational personnel that automatic action of engineered safety features and isolation signals shall not be manually overridden unless:

a. Continued operation of the engineered safety features or isolation signals will result in unsafe plant conditons, or

b. It is known or positively determined that the automatic action was initiated by a spurious or erroneous signal and it is verified that operaton of the engineered safety feature or isolation is not required, or

c. Approved procedures specifically allow manual override under specific conditions, and those conditions are verified to be satisfied.

Additionally, these points will be periodically restressed during the operator requalification training program.

SAD's have been revised to instruct operational personnel that when one or more confirmatory indicators are available, operational decisions shall not be made based solely on a single plant parameter indication. Additionally, the SAD provides instructions to operational personnel that all available information should be considered in decisions to manually initiate, terminate, or control operation of safety systems.

RESPONSE TO ITEM 1c

Attendance during the review described in 1a, above was documented.

ITEM 2

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 TO ITEM 2

Containment isolation design and procedures have been reviewed. Containment isolation of all lines whose isolation does not degrade needed safety features or cooling capability is initiated either automatically or manually upon automatic initiation of safety injection.

ITEM 3

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 TO ITEM 3

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

High Pressure Core Spray (HPCS) System
Reactor Core Isolation Cooling (RCIC) System
Safety Relief Valves (SRV) and Automatic Depressurization System (ADS)
Low Pressure Core Spray (LPCS) System
Low Pressure Coolant Injection (LPCI) Mode of the Residual Heat Removal (RHR) System

The operation of systems needed to achieve initial core cooling, containment cooling, and extended core cooling for long term plant shutdown is decribed below.

a. INITIAL CORE COOLING

Following loss of feedwater and subsequent reactor scram, a low reactor water level signal will automatically initiate main steam line isolation valve closure. The safety relief valves (SRV's) will automatically actuate to maintain reactor pressure. At the same time, the low water level signal automatically initiates the HPCS and RCIC Systems. These systems will continue to inject water into the reactor vessel until a high water level signal closes the HPCS injective valve and trips the RCIC system. Following a high reactor water level trip, the HPCS injection valve will again reopen when reactor water level decreases to the low water level setpoint. The RCIC System must be manually reset before it will reinitiate after a high water level trip. The HPCS and RCIC Systems have redundant supplies of water, normally taking suction from the cycled condensate storage tanks (CST's). The HPCS and RCIC Systems suction will automatically transfer from the CST to the suppression pool if the CST water is depleted or if the suppression pool water level increases to a high level.

The licensed operator can manually initiate the HPCS and RCIC Systems from the main control room before the low reactor water level 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. This would prevent a trip of the systems due to high water level. The operator can verify that these systems are delivering water to the reactor vessel by any or all of the listed methods.

1. Verifying reactor water level increases when systems initiate using redundant level indicators.

2. Verifying system flow rates using flow indicators in the control room.

3. Verifying system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow from the reactor.

The HPCS and RCIC Systems can maintain reactor wate level at full reactor pressure and until pressure decreases to where low pressure systems such as the LPCI Mode of the RHR or Low Pressure Core Spray (LPCS) can maintain reactor water level.

b. CONTAINMENT COOLING

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would initiate the suppression pool cooling mode of RHR. This mode of operation removes heat resulting from safety relief valve (SRV) discharge and/or RCIC exhaust to the suppression pool. This is accomplished by placing one loop of the RHR System in the suppression pool cooling mode; (RHR suction from and discharge to the suppression pool through one RHR heat exchanger.)

The operator verifies proper operation of the RHR System containment cooling function from the main control room by:

1. Verifying RHR and Service Water (WS) System flow using system control room flow indicators.

2. Verifying correct RHR and Service Water System flow paths using control room position indication of motor-operated valves.

3. Monitoring suppression pool water temperature.

Even though one loop of the RHR is in the Suppression pool cooling mode, core cooling is its primary function. Thus, if a high drywell pressure or low water level 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. In addition, the HPCS and LPCS Systems would automatically start upon receipt of all ECCS initiation signals(s). The HPCS System functions as described in response 3.a and immediately injects water into the reactor vessel to maintain reactor water inventory. The LPCS and LPCI Mode of RHR would inject water into the reactor vessel if reactor pressure is below the respective

system discharge pressures. Upon receipt of conincident low reactor water level and high drywell pressure signals, and after a two minute time delay, the Automatic Depressurization System (ADS) would relieve reactor pressure to allow the low pressure systems (LPCS & LPCI) to inject water into the vessel. (Also see responses 5.a and 5.b)

c. 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 LPCI mode of one RHR loop and places that loop in the shutdown cooling mode as follows:

i. Trip the selected RHR pump
ii. Close motor operated valves (MOV's) in the suppression pool suction and discharge lines of the selected loop.
iii. Open the RHR shutdown cooling suction and dishcarge MOV's
iv. Restart the selected RHR pump

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 containment cooling mode.

ITEM 4

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 TO ITEM 4

Description of the reactor vessel level automatic initiation for the safety systems is provided in Chapter 7.3 of the FSAR. The description of the vessel level indication for manual initiation of the safety systems is described in Chapter 7.5 of the FSAR. As described in response to item 1, SAD's have been revised to provide instructions to operating personnel to utilize all available information.

ITEM 5

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 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 TO ITEM 5a

Approved station operating and training procedures review accomplished by September 15, 1979 to ensure that appropriate instructions clearly specify that operating personnel do not override automatic actions of engineered safety features unless continued operation of these systems will result in unsafe plant conditions. However, it has been determined that several valid reasons exist for allowing an override of an automatic initiation signal or shutdown of a system after it has been automatically initialed (see response to Item 1, above). For example,

a. If an automatic initiation of the HPCS System and RCIC System occurs, the operator is permitted to shutdown the HPCS System if the RCIC System is capable of maintaining vessel level. This is allowed to prevent a trip of both systems due to high water level. As noted in response to Item 3 of this Bulletin, a trip of the RCIC System requires manual operator action to reset.

b. The procedures allow the operator to manually override automatic actuation of the ADS if it has been determined that adequate water level is being maintained by the HPCS System. In this case, LPCI or LPCS is not required and, therefore, ADS actuation can be interrupted. This override is permitted to allow a controlled cooldown and depressurization of the reactor and prevents injection of suppression pool water into the reactor when it is not required.

c. The procedures allow transfer of part or all of the RHR System from the LPCI mode of operation to the Suppression Pool Cooling or Shutdown Cooling modes of operation when adequate reactor water level is maintained with part of the RHR System and/or other systems. This is permitted to insure that suppression pool water temperature and containment pressure limits are maintained and provides controlled cooldown of the primary system.

RESPONSE TO ITEM 5b

The SAD concerning station operations has been revised as stated in the response to item 1 to assure operators consider all available information in decisions to take manual action. Operating procedures for specific events do describe expected parameter indications. Clarification, and where appropriate amplification will be made to specific operating procedures that describe parameter indications. Changes to procedures will be included in appropriate portions of operator training sessions. However, it should be recognized that events may occur such that vessel level indication might be the only immediately obvious parameter affected. We are reluctant to issue instructions which might be considered contrary to the directive for operators to believe and respond conservatively to instrument indications unless the indications are proven to be incorrect.

ITEM 6.

Review all safety-related valve positions, positioning requirements and

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controls to assure that valves remain positioned (open or closed) 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 TO ITEM 6

Prior to a plant startup safety related system valve checklists are completed to establish and verify valve positioning. On plant restarts following short outages when major maintenance is not performed, checklists may be completed only on systems where maintenance was performed.

In accordance with CG&E Company policies and job descriptions, qualified operating personnel are the only individuals who may position valves. These activities are performed under the direction of the licensed reactor op- rator (nuclear control operator) or the licensed senior reactor operator shift supervisor).

veillance activities are performed using individual procedures which have prepared in accordance with Station Administrative Directives governing eillance. These directives provide guidance in regard to procedural ration to ensure that systems are properly returned to service following llance testing.

related operational checklists and all surveillance procedures are ed by an independent person knowledgeable in the operation of the and by the Station Review Board. Final approval is by the Station ndent. In addition, all procedures including checklists, are re- least once every two years for required changes (or prior to use the case for special procedures) and following unusual transients.

to specific procedural control, licensed operators check main switches, annunciator panels, etc. for normal indications his includes valve position indications. Valves critical to ion but without position indication in the control room are ed in the required position. Their positioning is verified of startup checklists and in most cases with periodic com- eillance testing.

ities on safety-related equipment are performed using a est (WR). The Work Request form identifies isolation d and specifically identifies when the equipment is re- eration.

procedures, controls, and reviews described above are oper valve positioning.

odes and procedures for all systems designed to ioactive gases and liquids out of the primary t undesired pumping, venting or other release gases will not occur inadvertently.

ITEM 7 CONT'D

In particular, ensure that such an occurence 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 TO ITEM 7

All systems designed to transfer potentially radioactive gases and liquids from the primary containment are provided with automatic isolation valves. Isolation signals are initiated by a variety of reactor, containment or system conditions. The trip setpoints for these automatic isolation signals are listed in Technical Specification Table 3.3.2-2. Valve groups that are operated by these trips are listed in Technical Specification Table 3.3.2-1. These valve groups are listed in Technical Specification Table 3.6.3-1.

The containment radiation monitoring system is not part of the containment isolation system with the exception of the main steam line radiation monitors. The system provides information to the operator for the manual control of the primary containment systems.

ITEM 8

Review and modify as necessary your maintenance and test procedures to ensure that the 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 TO ITEM 8a, b and c

The removal of equipment from service is controlled administratively and/or with the use of the WR. The procedures which deal with equipment isolation specifically reference the responsible individuals to the applicable technical specification.

RESPONSE TO ITEM 8a, b and c CONT'D

Maintenance performed on safety-related equipment is controlled by the WR. The (Work Request) SAD assigns the Shift Supervisor the responsibility to identify technical specification requirements that pertain to any maintenance activity. Any post work surveillance testing that is required is also identified on the WR.

Written authorization to begin safety related corrective maintenance and any surveillance testing must be obtained from the Shift Supervisor. Removing any equipment from service must be reviewed and authorized by the Shift Supervisor. In addition, station operators are the only personnel who remove equipment from service and return it to service. Records of equipment tagged out and jumper and lifted leads are maintained by the operations group.

ITEM 9

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 TO ITEM 9

Prompt reporting procedures for NRC notification will be revised to establish a continuous open line of communication with the NRC as rapidly as possible in the event the reactor is not in a controlled or expected condition of operation. It is our intent to work with the Region III Office of Inspection and Enforcement in the development of continuous communication channels.

ITEM 10

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 TO ITEM 10

In the event hydrogen gas is generated by metal water reaction or radiolysis, the following methods are available to relieve the gas from the reactor vessel (primary system).

a. SAFETY RELIEF VALVES (SRV's)

There are thirteen SRV's located on the main steam lines that relieve to the quenchers located below the suppression pool water level. Since there is about 20 feet between the top of the core and the main steam line nozzles, a large volume of noncondensable gas can be relieved to the suppression pool via this pathway.

b. LOSS OF COOLANT ACCIDENT

A direct leakage path to the primary containment is created for release of noncondensables for certain postulated line ruptures.

c. REACTOR HEAD VENT

The reactor vessel head vent relieves directly from the top of the vessel head via remote manual control from the main control room. The vent is directly piped to the reactor building equipment drain tank, and under supervision of a licensed operator or senior operator, the MOV's can be operated to relieve noncondensable gas to the primary containment.

After venting the hydrogen gas from the reactor vessel to the primary containment, the condensation of hydrogen and oxygen is continuously monitored by redundant trains of containment monitors. The Primary Containment Combustable Gas Control System is activated remotely from the main control room, and this system, through the use of hydrogen recombiners, can adequately handle the postulated volume of hydrogen gas generated from radiolysis and/or metal water reaction.

Operating procedures addressing the generation of hydrogen gas in the reactor vessel and release of hydrogen gas to the primary containment will be reviewed by December 15, 1979.