GL 79-23

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

June 20, 1979

TO ALL BOILING WATER REACTOR LICENSEES

SUBJECT: NRC STAFF REVIEW OF RESPONSES TO I&E BULLETIN 79-08

The purpose of this letter is to advise you of the status of our review of licensee responses to I&E Bulletin 79-08 (IEB 79-08), and to identify related concerns which we believe require your further consideration.

The priority established for the review of I&E Bulletins related to the Three Mile Island, Unit 2 accident (TMI-2) is (1) B&W plants (2) C-E and Westinghouse plants, and (3) boiling water reactor plants. For this reason, as well as the limited resources within the NRC staff to apply to these reviews, we have now targeted for completion in August 1979. In the interim, we have scheduled a joint meeting at our Bethesda, Maryland office with all boiling water reactor licensees. This meeting is scheduled for 9:00 am on June 28, 1979, in Rooms P110/114 of our Phillips Building office in Bethesda, Maryland. You will be expected to attend the meeting and be prepared to discuss, among other things, those matters listed below along with a schedule and procedure for providing the information needed by the staff to complete the review of these issues.

(1) When our preliminary review of the responses to IEB 79-08 is complete, we will advise you of any items not satisfactorily resolved. In certain instances, this may include cases where licensee responses differ, without apparent justification, from General Electric Company (GE) recommendations. A copy of the GE recommendations is provided as Enclosure 1.

(2) We expect to prepare a generic report on (TMI-2) matters related to boiling water reactor operating plants. We expect that this report will recommend, among other things, further analyses of transients and small reactor coolant system breaks (including stuck-open relief valves), the development of appropriate written procedural guidance to operators as indicated by these analyses, and further training of operators in the use of these new procedures.

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(3) In certain instances, licensees are using fuel and relying on safety analyses which are not provided by the nuclear steam supply system (NSSS) designer. As a result, it is not clear to us what the respective roles of the

licensees, NSSS designers, fuel suppliers, and/or other parties should be in implementing any requirements that may evolve from item (2) above.

(4) The Advisory Committee on Reactor Safeguards (ACRS) has issued five letters to the Commission as a result of their examination of the TMI-2 accident. We believe that it would be mutually beneficial for each utility to provide specific comments on those ACRS recommendations having potential impact on plant design and/or operation. A summary of the ACRS recommendations is provided as Enclosure 2.

(5) Individual pressurized water reactor licensees have indicated an interest in meeting directly with the staff regarding the bulletin items for their facilities. Experience to date has demonstrated that the staff does not have time to meet individually with each licensee to resolve those items that are truly generic to all licensees.

It is clear that there are a significant number of technical issues yet to be resolved for a large number of operating plants. There are limited resources available within the staff to perform the necessary work. This situation is exacerbated by the need to conduct similar and concurrent activities with owners of pressurized water and boiling water operating plants. At the same time, there is a need to resolve these issues promptly.

In this regard, we believe there is a compelling need to establish an owner's group for boiling water reactor operating plants. We expect that such a group would be needed for the remainder of calendar year 1979. Owner's groups have worked effectively in the past in minimizing staff and industry resource requirements to resolve other generic problems. We strongly urge your to meet with other owners of boiling water reactor operating plants to consider the formation of such a group prior to the forthcoming generic meeting with the staff discussed above. This will be one of the principal agenda items at that meeting.

Please note that our investigation of a number of areas related to the TMI-2 accident, including the ACRS recommendations and the action items from NUREG-0560, will be included specifically as part of the staff's "Lessons Learned" activity. You can expect additional correspondence in the future on those items.

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If you require any clarification of the matters discussed in this letter, please contact William F. Kane, the staff's assigned project manager for these activities on boiling water reactor operating plants. Mr. Kane may be reached at 301-492-7745.

Sincerely,

Thomas A. Ippolito, Chief

Operating Reactors Branch No 3
Division of Operating Reactors

Enclosures:
As stated

cc w/enclosures:
Service List

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Enclosure 1
(GE Recommendations)

April 20, 1979

Subject: IE BULLETIN 79-08, EVENTS RELEVANT TO BOILING WATER REACTORS IDENTIFIED DURING THREE MILE ISLAND INCIDENT DATE APRIL 14, 1979

NRC Bulletin 79-08 issued to all BWR reactor facilities with operating licenses and requested responses to eleven questions. While most of the questions pertain to operator performance and plant procedures, several are related to the BWR to a loss of feedwater flow and should be helpful in responding to question 1 of the bulletin. The attached writeup is provided for your use in preparing a total response to questions 2,3,4,5b and 10 and is intended to assist BWR owners in preparing a consistent response to the bulletin. We suggest that BWR owners communicate with each other to assure consistency of their response.

The attachment provides the basis for our conclusion that the design of the BWR has already considered the concerns referenced in the NRC bulletin, such as:

(1) response to transient caused by equipment failures,
(2) methods of auxiliary heat removal,
(3) redundancy of reactor water level indication,
(4) containment isolation, and
(5) handling of hydrogen

Since the writeup was developed for a typical BWR-2, BWR-3, and BWR-4, some of the information may need to be modified to be directly applicable to your specific plant.

Should you have any additional questions, please contact this office.

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NRC QUESTION 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 SCOPE

This proposed response covers the review of the containment isolation design. The discussion covers initiation of the manual and automatic features of all lines whose isolation does not degrade needed safety features. BWR owners should include the NRC required evaluation of procedures or specific Balance of Plant systems.

RESPONSE APPROACH

A review was made of drawings and specifications for one Boiling Water Reactor (BWR) of each type (ie, BWR/4, BWR/3, and BWR/2). Items reviewed were General Electric Process and Instrument Diagrams (P&ID's), elementary diagrams and specifications for Nuclear Steam Supply System safety systems, Reactor Protection System (RSP), the Nuclear Boiler System, and other systems.

CONCLUSIONS

The BWR design provides containment and reactor coolant pressure boundary (RCPB) isolation (excluding emergency core cooling and make-up systems). The isolation occurs upon reactor vessel low water level or high drywell pressure prior to or simultaneous with initiation of emergency core cooling and safety injection systems. The isolation valves will remain closed until operator action is taken even if the initiating signal clears. The operator cannot shutoff Emergency Core Cooling Systems (ECCS) if the low water level condition exists; therefore, the containment and RCPB isolation is adequate and does not degrade required safety functions or core cooling capability and no changes are necessary. During initial plant startup containment isolation was tested prior to operation and is periodically retested in accordance with the plant Technical Specifications to assure continued operability.

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ADDITIONAL INFORMATION FOR NRC QUESTION 2

DISCUSSION

For convenience in providing the discussion of the isolation system, systems are separated and Group B systems represent Balance of Plant or Architect Engineer (AE) supplied systems. The isolation of each typical group is described separately.

SYSTEMS GROUPS DEFINITION

Category A - The Nuclear Steam Supply System (NSSS) includes the following four general groups of systems:

Group A1 - Non-safety systems which connect to the Reactor Coolant Pressure Boundary (RCPB), penetrate the containment and are not part of the Emergency Core Cooling Systems (ECCS) or other safety-related coolant injection systems. Examples of these non-safety systems are:
    Reactor Water Cleanup Systems (RWCU)
    Main Steam Lines

Feedwater Lines

Group A2 - Systems which connect to the RCPB and are part of the ECCS or safety related coolant injection system such as:

Reactor Core Isolation Coolant System (RCIC)
High Pressure Coolant Injection System (HPCI)
Low Pressure Core Spray System (LPCS)
Residual Heat Removal/Low Pressure Injection (RHR/LPCI)
Isolation Condensers
Automatic Depressurization System (ADS)

Group A3 - Systems which are shutdown cooling systems that are connected to RCPB and/or ECCS (eg. RHR shutdown cooling) but which are required to isolate so as to preserve the ECCS function.

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ADDITIONAL INFORMATION FOR NRC QUESTION 2

Group A4 - Systems which connect directly to containment, such as:

Drywell Floor Drain Pumping Systems
Equipment Drain Sump Pumping Systems
Containment Spray System

Category B - Systems which are outside General Electric scope of supply and connect directly to the containment consist of two general groups follows:

Group B1 - Containment ventilation systems or other Balance of Plant (BOP) Systems which connect to the containment atmosphere, including AE supplied Radwaste Systems.

Group B2 - Closed systems which penetrate the containment (e.g., closed cooling water, plant air, etc.) These systems do not connect to the RCPB or containment atmosphere.

ISOLATION FUNCTION

Piping penetrating the containment which do not degrade the core cooling capability are isolated prior to or simultaneous with the ECCS initiation in the following ways:

The Group A1 systems (except feedwater lines) automatically isolate at low reactor water level (level 2). This trip set point is the same one that automatically initiates the HPCI and RCIC systems. High drywell pressure also isolates this group of systems, except for the main steam lines.

Feedwater lines are automatically isolated with checkvalves. Feedwater can be manually isolated from the control room by remote manual switches. The reactor water cleanup (RWCU) system also isolates upon high temperature in the RWCU equipment spaces or on high differential flow in the RWCU lines. These isolation signals prevent loss of coolant. BWR/2, 3 and some BWR/4's isolate the RWCu at the higher level 3 - BWR owner needs to verify.

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ADDITIONAL INFORMATION FOR NRC QUESTION 2

Automatic isolation of the main steam lines is also initiated upon either 1) high temperatures in the steam tunnel or steam line spaces in the turbine building, 2) excess flow in any main steam line, 3) high radiation in the steam lines, 4) low steam line pressure, or 5) excess pressure in the main condenser.

THE GROUP A2 systems do not automatically isolate except for RCIC and HPCI which isolate upon excess steam flow or high temperature in their steam line or their equipment spaces.

THE GROUP A3 systems are automatically isolated by the low level set point (level 3) which is also the level at which the reactor scrams. Non-ECCS portions of the RHR system are also isolated by a high drywell pressure signal (2PSIG). This assures no loss of the LPCI function.

THE GROUP A4 systems isolate at either reactor water level 2 or upon high drywell pressure (2 PSIG).

THE GROUP B1 systems are automatically isolated at either low reactor water (level 2) or upon high drywell pressure. BWR owners should check implementation. High radiation instruments in the containment exhaust ducts are provided to isolate the containment ventilation system. BWR owners should check implementation.

THE GROUP B2 systems isolation is determined by the AE. Reactor low water level and high containment pressure signals are provided to the AE for this function. BWR owners should check for implementation.

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NRC QUESTION 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 SCOPE

This proposal response describes both automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems. These systems are used when the main feedwater system is not operable. The procedures are described in summary form assuming the reactor is scrammed and isolated from the main condenser.

RESPONSE APPROACH

GE has determined the necessary actions and procedures from design and operating procedures documents. BWR plant owners should check plant operating procedures for consistency.

CONCLUSION

Automatic action provides abundant makeup water to the core for initial cooling. Long term core and containment cooling can be provided with few manual actions. Information is available to the operator in the control room to assist him in taking the required manual actions. Information in the control room permits the operator to verify that the objective of these actions is being achieved.

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ADDITIONAL INFORMATION FOR NRC QUESTION 3

DISCUSSION

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

High Pressure Coolant Injection System (HPCI)
Reactor Core Isolation Coolant System (RCIC)
Low Pressure Core Spray System (LPCS)
Residual Heat Removal (RHR)

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

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 makeup injection mode. These systems will continue to inject water into the vessel until high water level signal (level 8) automatically trips the systems.

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 located in the main control room. This action would be taken when control room alarms indicate low CST or suppression pool high water level.

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ADDITIONAL INFORMATION FOR NRC QUESTION 3

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) Verifying reactor water level increases when systems initiate (see water level discussion in response to Question 4).

b) Verify systems flow using flow indicators in the control room.

c) Verify system flow is to the reactor by checking control room position indication of motor operator valves. This assures no diversion of system flow to the reactor.

Therefore, the HPCI and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as the Core Spray (CS) or Low Pressure Coolant Injection (LPCI) can maintain water level.

CONTAINMENT COOLING

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from safety relief valve (SRV) discharge to the suppression pool. This would be accomplished by placing the Residual Heat Removal (RHR) System in the containment (suppression pool) cooling mode, ie, RHR suction from and discharge to the suppression pool.

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

a) Verifying 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.

c) On branch lines that could divert flow from the required flow paths, close the motor-operated valves and not the effect on RHR and SW flow rate.

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ADDITIONAL INFORMATION FOR NRC QUESTION 3

Even though the RHR is in the containment cooling mode, core cooling is its primary function. Thus, if a high drywell pressure signal is received at any time during the period when the RHR is in the containment 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.

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 containment cooling mode of one of the RHR containment cooling loops and places the loop in the shutdown cooling mode as follows:

1. trip the RHR pumps,

2. close motor-operated valves in the suppression pool suction and discharge lines,

3. open suction valves from and discharge valves to the reactor vessel, and

4. restart the PHR pumps.

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

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NRC QUESTION 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 SCOPE

The proposed response to this question covers the uses and types of reactor vessel water level indication (or recording) for both automatic and manual initiation of Nuclear Steam Supply Systems (NSSS) safety sytems. Descriptions are also given of other instrumentation available to the operator for determining reactor water level. Further information on other available instrumentation is in the response to question 5b.

BWR plant owners should review operators' instructions and any Balance of Plant (BOP) instrumentation.

RESPONSE APPROACH

A review was made of drawings and specifications for one Boiling Water Reactor of each type (ie, BWR/4, BWR/3. and BWR/2). Items reviewed were General Electric Process and Instrument Diagrams (P&ID's), elementary diagrams and specifications for Nuclear Steam Supply System safety systems, Reactor Protection Systems (RPS) and the Nuclear Boiler Systems.

CONCLUSIONS

Reactor vessel water level in the BWR is continuously monitored by 7 (5 in BWR/2) indicators or recorders for normal, transient and accident conditions. Those monitors used to provide automatic safety equipment initiation are arranged in a redundant array with two instruments in each of two or more independent electronic divisions. Thus, adequate information is provided to automatically initiate safety actions and provide the operator with assurance of the vessel water level at all times.

These water level measurement devices have operated in BWR plant for 20 years. Test of BWR water level instrumentation under simulated steam and water line breaks have been conducted showing satisfactory performance.

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ADDITIONAL INFORMATION FOR NRC QUESTION 4

DISCUSSION

The range of reactor vessel water level from below the top of the active fuel area up to the top of the vessel is covered by a combination of narrow and wide-range instruments. Level is indicated and/or recorded in the control room. Typically, there are seven in BWR/3 and 4, and five BWR/2 plants. (see figure.)

A separate set of narrow-range level instrumentation on separate condensing chambers (typically five per plant) provides reactor level control via the reactor feedwater system. This set also indicates or records in the control room (typically three level indicators and one level recorder).
The safety-related systems or functions served by safety-related reactor water level instrumentation are:

Reactor Core Isolation Coolant System (RCIC)
High Pressure Coolant Injection System (HPCI)
Low Pressure Core Spray System (LPCS)
Residual Heat Removal/Low Pressure Injection (RHR/LPCI)
Automatic Depressurization System (ADS)
Nuclear Steam Supply Shutoff System (NS4)

All systems automatically initiate on low reactor water level. In addition, the RCIC and HPCI systems shutdown on high reactor vessel level. In all cases, except the RCIC, these systems automatically restart if low reactor level is again reached. (See response to questions 3 for further discussion on this.) In the case of RCIC, manual resetting is required if high reactor vessel water level is reached.

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NRC QUESTION 5

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

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

RESPONSE SCOPE

The proposed response identifies alternate instrumentation to provide the operator with information to take corrective actions in the event of loss of reactor coolant or other abnormal conditions.

RESPONSE APPROACH

In determining the proposed response, a review was made of drawings and specifications for one typical Boiling Water Reactor (BWR) of each type (ie, BWR/4, BWR/3, and BWR/2). Items reviewed were General Electric Process and Instrument Diagrams (P&ID's), and elementary diagrams and specifications.

CONCLUSIONS

Over a dozen other types of instrumentation in the BWR provide the operator with indirect indication of reactor vessel coolant inventory changes and could inform the operator of the need to take corrective actions.

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ADDITIONAL INFORMATION FOR NRC QUESTION 5

DISCUSSION

Some of the instrumentation which the operator can use to determine changes in reactor coolant inventory or other abnormal conditions are:

Drywell High Pressure
Drywell High Radioactivity Levels (owner check applicability)
Suppression Pool High Temperature
Safety Relief Valve (SRV) Discharge High Temperature
High Feedwater Flow Rates
High Main Steam Flow
High Containment and Equipment Area Temperatures
High Differential Flow-Reactor Water Cleanup System
Abnormal Reactor Pressure
High Suppression Pool Water Level
High Drywell and Containment Sump Fill And Pumpout Rate
Valve Stem Leakoff High Temperature

An example of the use of this additional information by the operator is a follows: Drywell high pressure is an indirect indication of coolant loss. Coincident high suppression pool temperature further verifies a loss of reactor coolant. High SRV discharge temperature would pinpoint loss of coolant via an open valve.

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

High Neutron Flux
High Process Monitor Radiation Levels
Main Turbine Status Instrumentation
Abnormal Reactor Recirculation Flow
High Electrical Current (Amperes) to Pump Motors

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NRC QUESTION 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 COVERAGE

The proposed response to this question discusses the system that are available for removing hydrogen from the primary system as well as treatment and control of hydrogen in the containment.

Procedures for these operations are unique to each plant and should be available addressed by each plant owner.

CONCLUSIONS

The BWR has design features to safely dispose of any significant amounts of hydrogen either in the primary system or in the primary containment.

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ADDITIONAL INFORMATION FOR NRC QUESTION 10

DISCUSSION

During normal operation, the Reactor Pressure Vessel dome is filled with steam, which flows to the turbine. During reactor isolation, the dome is automatically vented through the SRVs to the suppression pool. In addition, the Reactor Pressure Vessel head has a vent line with a valve remotely operated from the Control Room.

In the event of significant hydrogen release to the primary containment, the containment atmosphere dilution system maintains hydrogen below flammability. In addition, there are usually several other systems such as the containment atmospheric monitoring system, hydrogen recombiners, and containment purge via standby gas treatment to provide long term hydrogen control. For those plants with inerting, the possibility of hydrogen flammability is precluded.

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ENCLOSURE 2

ACRS RECOMMENDATIONS

A. Letter, M. Carbon to Chairman Hendrie, dated April 7, 1979

Recommendation 1* - Perform further analyses of small break transients and accidents.

Recommendation 2 - Provide operator additional information and means to follow the course of an accident; as a minimum, consider expeditiously:

(a)** unambiguous RV level indication

(b) remotely controlled vent for RCS high points

Recommendation 3 - Item 4b of Bulletin 79-05A considered unduly prescriptive in view of uncertainties in predicting course of anomalous small break transients/accidents.

B. Letter, R. Farley to Commissioners, dated April 18, 1979

Recommendation 1 - Natural Circulation-related Items

a.* Detailed analyses of natural circulation mode, supported as required by experiment, by licensees and NSSS vendors.

b.* Develop procedures for initiating natural circulation.

c.* Provide operator means for assurance that natural circulation has been established, e.g., by installation of instructions to indicate flow at low velocities.

d.* Expeditiously survey operating PWR's to determine whether suitable arrangements of PZR heaters and reliable on-site power distribution can be provided to assure this vital aspect of natural circulation capability.

e.* Operator should be adequately informed concerning RCS conditions which affect natural circulation capability, e.g.,

(1) indication that RCS is approaching

saturation condition by suitable display
to operator of Tc & Th and PZR pressure in
conjunction with steam tables

(2) use of flow exit temperature indicator by
fuel assembly thermocouples, where
available.

* Amplified in Interim Report #2
** Amplified in Interim Report #3
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Enclosure - page 2

ACRS Recommendations

Recommendation 2*- Thermocouples used to measure fuel assembly exit temperatures to determine core performance should be used where currently available, to guide operator concerning core status (full range capability).

Recommendation 3 - Operating reactors should be given priority regarding definition and implementation of instrumentation to diagnose and follow the course of a serious accident, including

(a)* improved sampling procedures under accident conditons

(b)** improved techniques to provide guidance to offsite authorities.

Recommendations 4 - Reiterates previous recommendations that high priority be given to "research to improve reactor safety"

(a)* research on behavior of LWRs during anomalous transients

(b)* NRC to develop capability to simulate wide range of postulated transients and accident conditions.

Recommendation 5 - Consideration should be given to additional monitoring of ESF equipment status, and to supporting services, to help assure availability at all times.

C. Letter, M. Carbon to Acting Chairman Gilinsky dated April 20, 1979

Recommendation 1 - Initiate immediately a survey of operating procedures for achieving natural circulation, including:

(a)* event involving loss of offsite power (b)* consideration of role of PZR heaters.

* Amplified in Interim Report No 2 dated 5/15/79
** Amplified in Interim Report No 3 dated 5/15/79

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D. Interim Report No. 3 dated May 16, 1979

Recommendation 1 - Examine operator qualifications, training and licensing, and requalificatin training and testing.

Recommendation 2 - Establish formal procedures for the use of LER information:

(a) in training supervisory and maintenance personnel

(b) in licensing and requalification of plant operating personnel

(c) in anticipating safety problems

Recommendation 3 - Consider formal review of operating procedures for severe transients by inter-disciplinary team, and develop more standardized formats for such procedures.

Recommendation 4 - Re-examine comprehensively the adequacy of design, testing and maintenance of offsite and onsite AC and DC power supplies with emphasis on:

(a) failure mode & effects analyses
(b) more systematic testing of power system reliability
(c) improved quality assurance and status monitoring of power supply systems

Recommendation 5 - Make a detailed evaluation of current capability to withstand station blackout, including:

(a) examination of natural circulation capability under such circumstances
(b) continuing availability of components needed for long-term cooling under such circumstances
(c) potential for improvement in capability to survive extended blackout

Recommendation 6 - Examine a wide range of anomalous transients and degraded accidents which might lead to water hammer, with emphasis on:

(a) controlling or preventing such conditions
(b) research to provide a better basis for control or prevention of such conditions

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Recommendation 7 - Plan and define NRC role in emergencies, including consideration of:

(a) assurance that formal emergency plans, procedures and organizations are in place
(b) designation of emergency technical advisory teams (names and alternates)
(c) compilation of an inventory of equipment and materials needed in unusual conditions or situations

Recommendation 8 - Review and revise within three months:

(a) licensees' bases for obtaining offsite advise and assistance in emergencies from within and outside company
(b) licensees' current bases for notifying and providing information to offsite authorities in emergencies

Recommendation 9 - Examine the lessons learned at TMI-2, including consideration of the following:

(a) behavior, failure modes, survivability and other aspects of TMI-2 components and systems as part of the long-term recovery process
(b) determine if design changes are necessary to facilitate decontamination and recovery of major nuclear power plant systems

Recommendation 10- Expedite resolution of unresolved safety issues by the following means:

(a) suitable studies on a timely basis by licensees to augment NRC staff efforts
(b) use of consultant and contractor support by NRC staff

Recommendation 11- Augment expeditiously the NRC staff capability to deal with problems in reactor and fuel cycle chemistry in the following areas:

(a) behavior of PWR & BWR coolants and other materials under radiation conditions
(b) generation, handling & disposal of radiolytic (or other) H2 at nuclear facilities

(c) performance of chemical additives in containment sprays
(d) processing and disposal techniques for high and low level radioactive wastes

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(e) chemical operations in other parts of nuclear fuel cycle
(f) chemical treatment operations involved in recovery, decontamination or decommissioning of nuclear facilities

Recommendation 12- Reconsider whether or not use of the Single Failure Criterion establishes an appropriate level of reliability for reactor safety systems

Recommendation 13- With respect to safety research:

(a) consideration should be given to augmentation of the FY80 NRC safety research budget
(b) consider orienting a larger part of the safety research budget toward exploratory (as opposed to confirmatory) research

Recommendation 14- Perform design studies of a filtered venting or purging option for containments for possible use in the event of a serious accident

E. Interim Report No 2, dated May 16, 1979

Amplified many of the recommendations included in earlier ACRS letters dated April 7, April 18, and April 20, 1979, including ACRS views on relative priorities to be assigned a number of those earlier recommendations. (Address amplifications and suggested priority assignments as appropriate.)