GENERAL ELECTRIC

EVALUATION OF THE

NEED FOR

BWR CORE THERMOCOUPLES

DJR:ggo/214L1

INDEX

DJR:ggo/214L2
7/6/81

ABSTRACT

This report provides an evaluation of the need for incorporation of incore thermocouples in the BWR.

Regulatory documents including NUREG-0737 and Regulatory Guide 1.97 Revision 2 have suggested that the addition of this instrumentation might improve BWR post-accident monitoring capability.

An evaluation is provided which concludes that core thermocouples are inappropriate when considering

1) they do not provide useful additional information,

2) incorporation would subject plant personnel to increased radiation exposure,

3) there is no reduction in public risk.

1.0 Introduction

As a result of the Three Mile Island - 2 accident, the NRC perceived a need for all reactor types to use in-core thermocouples to provide an unambiguous, easy-to-interpret indication of inadequate core cooling. The basis for this new requirement was that in-core thermocouples were of some value in determining core conditions following that accident. This position was articulated through Item II.F.2 of NUREG-0737.

Approximately concurrent with the NUREG-0737 requirement, Regulatory Guide 1.97 Revision 2, Draft 2 was issued. This draft also required In-Core Thermocouples for BWR's. The reason cited for requiring them for post-accident monitoring (the topic of the Regulatory Guide) was to indicate the potential for, or actual occurrence of, fuel cladding breach. The NRC staff also indicated that they desired to identify local hot areas and the propagation of core damage. They suggested that approximately 50 thermocouples should be utilized. This quantity was judged by the NRC to be sufficient to detect blockage of 5-10% of the core with no spray (or other ECCS) at a high confidence level and with a sufficient allowance for attrition.

GE and the BWR Owner's Group subsequently developed arguments (Reference 1) which eventually led to the current version (Revision 2) of Regulatory Guide 1.97. In this version the NRC staff cites its reasons for now requiring 16 in-core thermocouples for BWR's:

1) monitor core cooling, and

2) provide a diverse indication of water level.

BWR's do not need in-core thermocouples for any purpose. General Electric has nonetheless evaluated the use of such devices in the BWR. The most practical location to install thermocouples in a BWR is in the in-core power range monitor (PRM) instrument assemblies. All other locations would require additional penetrations and major redesign of the vessel internals and/or the fuel bundles. An analysis of the temperature response of a thermocouple in the PRM assembly indicates that it would only provide an indication of gross core discharge superheat conditions in the highly unlikely event that no water makeup systems were operating for an extended period. For such a situation, a single thermocouple anywhere above the core would provide comparably useful information as to the existence of a bulk super heat condition.

It is General Electric's concern that the installation of in-core thermocouples in the BWR will not satisfy either NRC purpose except for one narrow case of very low probability as described later in this report. Even for this case thermocouples would provide information of no practical use to either the operators or the on-site engineering staff.

This installation of instrumentation is costly in dollars but more significantly also in Man-Rem dosage during installation and maintenance. Installation of in-core thermocouples in BWR's is inconsistent with the ALARA and cost-benefit concepts.

General Electric believes the solution to achieving adequate post accident monitoring of core cooling and reactor water level is through:

1) present BWR water level instrumentation, and

2) other instrumentation required by Regulatory Guide 1.97 or already incorporated into BWR plant designs which provide diverse indication of water level for the same narrow case as thermocouples.

2.0 Operational Evaluation of Thermocouples in BWR's

Monitoring of Core Cooling and providing Diverse Indication of Water Level are postulated reasons for Incore Thermocouples in BWR systems. These functions are not performed by Incore Thermocouples. A situational analysis for this conclusion follows.

2.1 Monitor Core Cooling

In assessing the plant safety improvement resulting from core exit temperature measurements, several periods during the course of an event must be evaluated.

Prior to Core Uncovery

The first period is prior to core uncovery. The BWR operates under saturated conditions with very strong natural circulation inside the reactor pressure vessel. Studies (Reference 2) have shown that, as long as the core remains covered with water, adequate core cooling is assured. Therefore cladding breach must be preceded by a challenge threatening to uncover the core.

Because of this inherent feature of the BWR, water level is also the primary measure of accomplishment of the core cooling safety function during accident situations. Thus, reactor water level is the key parameter for BWR's on which both automatic and operator actions are based.

The BWR has highly reliable and redundant reactor water level measuring systems. BWR's also are equipped with symptom based inventory control emergency procedure guidelines which direct the operator to utilize the multiple coolant injection and spray systems. A brief discussion of BWR Water Level Measurement is provided in Appendix A.

The BWR is designed with a multiplicity of water sources and injection delivery systems to maintain the core covered. These systems, together with the inherent BWR natural circulation, provide decay heat removal capability. Operational performance of these systems during transients and accidents is presented in Reference 2.

The BWR has multiple sources of water injection to the reactor core from below (flooding). This capability for core flooding is established by both high pressure systems and low pressure systems. Not only is there redundancy of systems but also redundancy of pumping capability within the systems.

In addition to flooding systems, which provide water to the reactor core from below, typically there are also the High Pressure Core Spray (HPCS) and/or Low Pressure Core Spray (LPCS) systems. These systems supply water spray directly onto the core from above. For a no-break situation (scram, reactor isolation, or stuck-open relief valve), about only 300-gpm injection is required to keep the core from ever being uncovered. There are a total of 13-15 pumps which can supply spray and flooding to the core.

During this time period, core exit thermocouples would be indicating, at most, saturation temperature corresponding to the reactor vessel pressure. Core exit thermocouple readings would be erratically indicating lower temperatures due to the subcooling effect of ECCS (core spray and LPCI). The use of core exit thermocouples would not satisfy the purpose of monitoring core cooling and would not provide useful additional information for the plant operator and the erratic readings may be confusing.

Following Core Uncovery

The second time period when knowledge of core exit temperatures might be useful is during fuel heatup following core uncovery. It is during this time that the potential for cladding breach exists, and, depending on the duration and amount of core uncovery, the potential exists for creating local flow blockage as a result of core damage. Reactor vessel water level provides the ability to detect core uncovery and, thus, by itself, indicates the potential for cladding perforations. Automatic and operator manual actions would already be underway to restore water level to cover the core. BWR's are uniquely provided with symptom based emergency guidelines for reactor inventory control. Continued monitoring of reactor water level and water makeup system performance parameters provides the capability for monitoring core cooling.

In addition there are many other parameters available to the operator that are reliable indicators of actual fuel clad breach, and thus inadequate core cooling. These include high steam line radiation, high offgas radiation levels, high area radiation levels in the containment, high hydrogen concentration in the containment, and high radioactivity in reactor or suppression pool water. Details of these provisions are discussed in Appendix B.

Core exit temperature measurement will not provide an unambiguous indication of either the potential for or actual clad damage. This results since the BWR's multiple, safety-grade core spray systems would continue to supply water spray over the top of the core even though the core may be uncovered in a bulk sense. Even if there is only one core spray system functioning (one of two provided), the core exit temperature, whether measured locally or in bulk, will not be superheated. The core sprays need only provide 300 gallons per minute of their total typical design flow rate of 12,000 gpm to remove any superheating in the steam. In the BWR 5 and 6 designs, the Low Pressure Coolant Injection (LPCI) system directly floods the core bypass region providing further subcooling.

In summary, during fuel heat-up following core uncovery, there is only one condition for the BWR that core exit temperature measurement would provide unambiguous and definitive information. This occurs in the highly unlikely event that, following a loss of water inventory, no normal, emergency, or alternate water makeup systems are available to replenish coolant inventory to the pressure vessel. During this narrow case the core is cooled by water and steam flow for a considerable period of time until the water in the core region is boiled off. Under such conditions, measurement of steam superheat anywhere above the core region would indicate core heatup and a low water level. However, should this condition occur the operator would be taking all appropriate actions to restore water level above the core based only on knowledge that water level is low and no injection is available. The operator guidelines even provide safe actions if no reactor water level indication is available and/or no injection is available.

Recovery Phase

The third time period, called the recovery phase, covers the interval after the operator has restored the water level in the core region. If there were no significant core damage, core exit temperature measurement would not provide any relevant information. The possibility of thermocouples providing useful information for operator actions was raised by the Staff in Draft 2 of Revision 2 of R.G. 1.97 for the situation when 5-10% of the core is damaged. The Staff contended that high core exit temperature readings would indicate localized propagating core damage (PCD) and guide the operator in long term decision making.

This position is unreasonable because: (a) once water level is restored in the core, core damage will not propagate to the rest of the core from the postulated 5-10% damaged core, and (b) temperature readings would not provide relevant information. Detection of PCD does not appear in the version of Revision 2 of R.G. 1.97, however, a discussion of this subject is included in Appendix B for completeness.

2.2 Diverse Indication of Water Level

In the BWR, water level is the primary measure of accomplishment of core cooling during accident situations. The requirement for diverse indication of water level was added to Revision 2 of R.G. 1.97 as a rationale for the use of thermocouples in BWR's. Appendix A provides a brief discussion of BWR reactor water level measurement systems. These systems are adequate and sufficient to reliably monitor water level during all inventory threatening events.

Water Level Relative to Core Uncovery

Since adequate core cooling and water level are closely tied in the BWR the discussion in Section 2.1 regarding the value of in-core thermocouples at various water levels appropriately applied here also. Only for the same narrow case of loss of inventory with no normal, emergency, or alternate water make up systems available would the thermocouples indicate core superheat and thus coolant level below the top of the core.

During this event the BWR not only has other diverse indication of such water level but also unique sumptom based proceedures to direct the operator whether he does or does not have water level indication.

Comparisons were made of core bypass thermocouple response to other instrumentation available to the operator (and required by R.G. 1.97) for a degraded loss of inventory event (all systems failed). This comparison shows that thermocouples would indicate core super heat (water level below the top of the core) approximately 10 minutes prior to detection of hydrogen production for the slowest event. This case involved a reactor isolation on loss of feedwater with no injection. Reactor inventory was slowly lost through intermittent SRV operation. For more challenging inventory threatening events such as a small break LOCA, this early warning detection by thermocouples could be reduced to only 2-3 minutes. Such time differences do not justify thermocouple installation.

Operator Response

Not only does the BWR operator have other instrumentation for diverse water level detection of this narrow case, but also symptom based emergency procedure guidelines that provide safe actions for:

1) no reactor water level indication available, and

2) reactor water level control with no systems available.

These procedures have been accepted by the NRC for implementation on all Near Term Operating License Plants.

With the highly reliable redundant water level monitoring system and with the emergency operating guidelines, the operator does not need in-core thermocouple information. In fact such new information may only confuse the operator by adding other parameters which would take his time away from restoring water level and may indicate erroneously.

3.0 Engineering and Maintenance Considerations

If incore thermocouples were to be incorporated, there are three possible locations within a BWR. These are: within or on the fuel assembly; on the shroud head with leads projecting downward to near the fuel assembly discharge; and in the PRM assemblies. The first alternate is considered unacceptable since it would create localized flow disturbances and cladding stress concentrations with the potential for initiating fuel damage. Both the first and second alternatives are also considered unacceptable due to the interference created between the thermocouple lead supports and the ECCS function - specifically core spray. They create an extremely difficult vessel and vessel internal design problem because of the multiple penetrations required in order to route the thermocouple leads. These alternatives could significantly impact the duration of each refueling outage.

Only placement in the PRM assemblies is technically feasible without extensive plant redesign. The PRM assembly is inserted into the reactor vessel from above the core with the vessel head and separator and dryer assemblies removed in earlier BWR design, and from below the core in the BWR 6 design.

In both BWR/6 and pre-BWR/6 designs, the PRM assemblies are secured to the top grid within the vessel. The top of the PRM latches approximately 10 inches below the top of the channel of the fuel assembly. The PRM latching mechanism design precludes locating the thermocouple higher than approximately 13 inches below the top of the fuel channel.

To withstand post-accident drywell environment of radiation, spray, immersion for BWR/6, requires metal-sheathed cabling with waterproof connectors from the vessel through the containment penetration. Based on preliminary design considerations, a minimum of two connectors, one located at the bottom of the other about one or two feet below the in-core housing flange, would be required for each PRM to permit its replacement. Difficulties are expected during both maintenance and installation.

Also, additional personnel exposure can be expected as a result of increased control rod drive (CRD) removal complexity. The presence of the thermocouple leads would further restrict personnel space availability and increase the possibility of damage to the cable leads and connectors during drive removal and replacement.

The total annual plant personnel exposure increase due to PRM, thermocouple and control rod drive maintenance would be in the range of 2 to 15 man-rems/ year for pre-BWR/6 plants and 1.7 to 14 man-rems/year for BWR/6 plants.

For installation, thermocouple leads would require routing from under the vessel in four separate arrays of about ten leads each, with the thermocouple leads distributed inside the pedestal in such a manner that each bundle would contain leads from the thermocouples located in each core quadrant. Complete isolation of these leads from the consequences of a specific accident is not feasible in operating plants, and is also thought to be unfeasible for plants under construction and design. Each of the four bundles of thermocouple leads is assumed to be routed through the containment in a structural housing to provide some protection during the accident (e.g., jet impingement), assuming two penetrations can be made available through which the thermocouple leads could be brought through containment, the installation of the leads in the containment is expected to take about 2,000 installation manhours. It should be noted that spare penetrations may not be available on operating plants considering other current NRC requirements. Including installation, modification engineering, and field engineering, the cost is approximately \$300,000* per plant.

Installation outside the drywell is assumed to be in a two-bundle configuration, with Division I power to one bundle and Division II power to the other bundle. Four multi-point recorders in the control room are assumed, although this is uncertain considering that the readings may be significantly delayed and illegible (due to similarity of readout).

On this basis, total installation cost is estimated to average \$600,000* on operating plants and \$400,000* on plants in construction. Exposures to installation personnel in each operating plant is estimated to be 100* man-rems assuming a 50 mr/hr general radiation field.

Note, application of the single-failure criterion of Table 1, Item 2 of R.G. 1.97 would eliminate readings from 50% of the thermocouples and accident consequence criteria could eliminate readings from another 25%. This presumed loss of installed thermocouples is of little consequence, since as previously discussed, exit thermocouples will be of little use in detecting local fuel temperature. Only 25% of the thermocouples (assuming 50 total) would still indicate bulk core uncovery with no water makup. Even this function is of little value, but at least in this sense, it is concluded that the single failure criterion can be met.

*These estimates are approximate. Precise definition would require plant by plant assessment. Probable accuracy: +50%

4.0 Risk Reduction Considerations

The nuclear industry has employed probabilistic risk assessment techniques for several years commencing with WASH-1400. Subsequent to the Three Mile Island-2 accident more emphasis have been placed on these techniques. GE has been involved in risk analysis since the issuance of WASH-1400 and has most recently completed an assessment of a BWR/4. Through this experience GE has gained extensive knowledge of design parameters for which the overall risk curve is sensitive.

GE has evaluated the usefulness of in-core thermocouples instrumentation relative to overall risk reduction. The narrow case during which thermocouples can work in the BWR (see Section 2.1) results in at least partial core uncovery. For risk assessment analysis this case would result in some level of core damage, the most conservative assumption being complete core melt.

The evaluated probability of core melt generally vary between 10^{-4} to 10^{-5} per reactor year. Present risk assessment techniques do not allow distinquishing between partial and complete core melt. It is judged that the frequency of partial core melt would be closer to the 10^{-4} end of the range, and thermocouples would not work following a complete core melt anyway. Therefore, installation of in-core thermocouples in the BWR would allow a few minute early warning (see Section 2.2) core uncovery and potential partial core melt at a frequency of 10^{-4} per reactor year.

The risk from core melt sequences is not dominated nor even mildly influenced by the operators early warning knowledge of incipient core damage. The risk assessments will allow the operator to restart available equipment; however, as already discussed thermocouples will not provide risk aid in this operation, only possible confusion. Therefore, risk assessments would show that installation of thermocouples will not reduce the probability of core melt or the risk to the public.

The industry is presently developing a safety goal for use in plant risk assessment. Several safety goals have been proposed and many of them include a cost-benefit criteria to preclude unnecessary cost for insignificant reduction in risk. In general this criterion is $100/man-rem.

As discussed in Section 3 radiation dose to plant personnel to install in-core thermocouples in BWR's is about 100 man-rem per plant with about 10 man-rem per year maintenance dosage per plant. Also discussed in Section 3 are cost estimates for installation of thermocouples of about $600,000.

Based on these estimates of cost and radiation exposure the cost benefit criterion comparison is $6,000/man-rem versus $100/man-rem, and this involves instrumentation of minimal or no usefulness to plant safety.

5.0 Conclusion

It has been demonstrated that core exit thermocouples provide no useful additional information to the operator. Moreover, the only practical location for their installation in any plant (operating or in design) would result in no significant enhancement of the operator's ability to protect the plant or public.

The combination of existing or planned (as a result of R.G. 1.97) instrumentation is sufficient to detect not only inadequate core cooling, but also monitor reactor water level diversely without in-core thermocouples. This is true for all possible loss of primary system coolant events independent of ECCS operational combinations. Detection of inadequate core cooling is expected to occur within a few minutes following any detection time by thermocouples. The introduction of thermocouples in the BWR constitutes not only a significant and costly design problem, but also subjects plant personnel to increased radiation exposure and provides no public risk improvement.

APPENDIX A.

BWR REACTOR WATER LEVEL MEASUREMENT

The nuclear fuel is contained within the reactor pressure vessel (RPV). The measurement and control of water level in the RPV is important to maintaining the effectiveness of the fuel as a barrier to fission product release. The BWR direct cycle design eliminates the need to monitor water level in multiple vessels since it is monitored in the RPV. Thus, the measurement is directly in the vessel containing the nuclear fuel.

Measurement is continuous and utilizes simple differential pressure concepts. Measurement instruments are calibrated for operating conditions, thus eliminating the need for complicated compensation techniques.

The design feature of a single vessel with direct measurement and abundant instrumentation results in operational simplicity. The operator need not be concerned with multiple variables to ensure knowledge of the level of water in the RPV. He need only monitor the single variable. Multiple reactor level indications are displayed in the control room. There are typically 29 remote-mounted differential pressure cells which transmit reactor water level signals to indicators (or recorders). The reactor water level indications are provided at the reactor control console or nearby panels in full view of the reactor operator. The measurement range is from near the steam separators to just above the top of the active fuel.

Safety control and information functions typically provided by the RPV level instruments include scram, containment isolation, Emergency Core Cooling System (ECCS) initiation, Reactor Core Isolation Cooling (RCIC) system initiation, Automatic Depressurization System (ADS) initiation, feedwater control, recirculation pump shutoff, Main Steam Isolation Valve (MSIV) closure and level readout and alarm functions in the control room for normal and post-accident conditions. Typically twenty-two indicating trip units provide wide-range and narrow-range reactor level safety-related trip signals. Each indicating trip unit provides level indication on back panels in the control room.

Because of these features, the BWR reactor water level measurement will perform satisfactorily for all modes of normal operation, anticipated transients and credible accident conditions. Detailed discussion of reactor water level measurement is presented in Reference 2.

RH:rf:ggt/214J2

APPENDIX B.

DETECTION OF PROPAGATING CORE DAMAGE

Core damage propagation, when the core is covered, has been discussed in a GE Licensing topical report (Reference 3). Because each bundle in the BWR core is surrounded by a flow channel, cross-flow between bundles is eliminated and any thermal-hydraulic effects of localized core damage remain localized. Each channel forms an essentially independent flow path connecting the upper and lower plenum and the core bypass region. To assure no damage to an undamaged fuel assembly, less than one gallon of coolant per minute must be provided. Since there are three independent flow paths into each fuel assembly (the top and bottom of the fuel bundle, and the flow paths between the bundle and bypass), any core damage propagation must start by almost complete blockage of all these paths. Calculations have been performed which show that all three paths have to be greater than 99% blocked for any damage to result. Even if almost total flow blockage of the bundles were postulated, this situation would not be likely to persist for long. Localized heating of the cladding would result in molten cladding coming in contact with the channel wall. Such localized heating of the channel would eventually form a hole in the channel, thus opening another flow path for the coolant from the bypass region to enter and cool the fuel rods.

Calculations have also been performed for the situation with 5-10% core damage and with an uncoolable geometry postulated to determine if superheated steam can be detected in the region around the damaged portion of the core. The calculations were done assuming the available instruments were those directly adjacent to the bundles in the damaged core region. The analyses show that the heat generation (decay heat and heat from metal water reaction) in the post-recovery phase are so low that, under all situations analyzed, nucleate boiling would be maintained and no superheat would be measured in the bypass region surrounding the damaged core.

It has been suggested by the NRC staff that if a temperature sensor was located adjacent to the assumed local blockage and if it were postulated tnat it could indicate some superheat, the operator could restart recirculation pumps. This would then force coolant through the partially blocked flow paths. However, as indicated above superheat would not be observed and the operator would have no knowledge that this action is necessary. In addition, because of the strong inherent natural circulation in the BWR, this action would be likely to be helpful for only a very limited situation where greater than 99% but less than 100% of all available flow paths were blocked. Therefore, operator actions would be no different: the principal emphasis would still be only on maintaining reactor water inventory. The addition of thermocouple data readouts may, indeed, add to operator confusion such that the reliability of operator action is reduced.

For the worst-case assumptions (i.e. uncovered core and no make-up) for which the NRC staff proposes that thermocouple indication would be useful, alternate means are available to provide trend information relating to the possible propagation of core damage (PCD). Those means which were previously available or are presently required by R.G. 1.97 and NUREG-0737 and provide direct indication of PCD, with or without ECCS functional, include: (1) reactor and suppression pool water/ containment air sampling and analysis for radioactive material, (@) containment gross gamma monitoring, and (#) containment hydrogen monitoring. Other measured variables required in R.G. 1.97 could also be used to infer PCD.

Analysis of reactor water samples would measure fission product activity and the concentration of dissolved hydrogen in the reactor water. The fission product activity from the gap/plenum would be released within several minutes after the onset of fuel clad perforations. It is expected that the reactor water sampling system will be sufficiently sensitive to detect the hydrogen concentration resulting from the reaction of as little as four pounds of zirconium. This is equivalent to a metal-water reaction involving about 3% of the cladding of a single fuel bundle.

For the dry-core case, vessel depressurization is expected. It will occur naturally if the event is initiated by a primary system break of sufficient size. It will occur by automatic or manual actuation for the no-break or small-break case because of safety/relief valve (S/RV) actuation. Thus, for the entire spectrum of initiating events, indication of core damage will be provided by various instruments in the containment. These include the suppression pool water/containment air sampler system, gross containment gamma monitor, and the containment hydrogen monitor. The gross gamma monitor would detect fuel clad gap/plenum activity release within several minutes from the onset of clad perforation.

Activity due to noble gases alone should provide sufficient indication of PCD. For the relatively straightforward case involving blockage of a single fuel assembly during normal plant operation, analysis (Reference 2) shows that within 9 seconds, fuel element melting would be detected by the steam line radiation monitor; scram and steamline closure would follow within 4 seconds. The offgas radiation monitor would alarm within two minutes.

The more complex case involving main steam isolation valve (MSIV) closure for reasons other than high steam line radiation has also been investigated. For this case, the safety relief valves (S/RV) open within seconds to relieve vessel pressure, and noble gases are transported via the S/RV discharge piping to the suppression pool water, then released to the containment free volume. The results of this analysis are illustrated in Figure 1 for the situation in which all reactor water makeup systems (normal, emergency and alternate) are postulated to remain inoperative for an extended period. Eventually the water level is reduced such that the readings on all thermocouples would increase with a distribution related to the core power distribution. For the situation in which the bulk water level has been significantly reduced there would be little or no correlation between thermocouple readings and core area cross sectional blockage. In this case the insufficient reactor water inventory would affect all fuel assemblies independent of whether or not blockage exists. The extent to which actual fuel failures occur could only be assessed by monitoring fission gas release to the primary system or the containment. Gross gamma monitoring should provide a more rapid indication of PCD for purposes of operator action. Confirming indications of the rate of PCD will be provided by the suppression pool water/containment air sampler system, as well as the containment hydrogen monitor. The containment hydrogen monitor is expected to be sufficiently sensitive to detect PCD as low as 2 to 3% core-wide, metal-water reaction per day.

RH:rf:ggt/214K4

REFERENCES

1. GE letter R. H. Buchholz to ACRS, "Regulatory Guide 1.97 (Draft 2 of Revision 2) - BWR Comments," August 4, 1980

2. NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," December 1980

3. NEDO-10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," October 1977

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

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

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)

Dupe # 8002220253

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.

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

FC-8935

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.