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Carolina Power & Light Company

October 8, 1980

U.S. Nuclear Regulatory Commission
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, DC 20555

Attention: D. G. Eisenhut, Director

Gentlemen:

SUBJECT: Preliminary Clarification of TMI Action Plan Requirements - BWR Owners' Group Comments

REFERENCES:

1) Letter, D. G. Eisenhut to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Preliminary Clarification of TMI Action Plan Requirements, Sep mber 5, 1980
2) Letter, M. S. Plesset to W. J. Dircks, gulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Access Plant and Environs Conditions During And Following An Accident," August 13, 1980
3) Letter, D. B. Waters to R. H. Vollmer, "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves," September 17, 1980
4) Letter, E. E. Utley to S. L. Ramos, "Comments on NUREG 0696 - Functional Criteria For Emergency Response Facilities," September 26, 1980
5) Letter, R. H. Buchholz to D. G. Eisenhut, "BWR Owners' Group Positions on NUREG-0578 Requirements," October 16, 1979

This letter summarizes the generic comments of the BWR Owners' Group on D. G. Eisenhut's September 5, 1980 letter titled "Preliminary Clarification of TMI Action Plan Requirements" (Reference 1). Many of these comments were provided verbally during the regional meetings held the week of September 22, 1980 and are repeated here for your convenience. This letter also identifies several new comments that have been identified since the regional meetings.

While this letter documents comments applicable to all utilities in the BWR Owners' Group, many utilities have identified additional comments or implementation problems unique to their own plants. These plant unique comments are being transmitted to the NRC to supplement this generic information via separate submittals.

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We hope that this information will be helpful to you in your revision of Reference 1. If you have further questions regarding the above comments, please contact Mr. S. J. Stark at General Electric (408) 925-1822.

Very truly yours,

David B. Waters

D. B. Waters, Chairman
BWR Owners' Group

DBW:SJS:na

Attachment

cc: J. A. Olshinski
P. W. Marriott
J. F. Schilder
BWR Owners' Group

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Requirement I.C.1

Guidance for the Evaluation and Development of Procedures for Accidents and Transients

Clarification of Implementation Requirement

Reanlaysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures should be completed and submitted to the NRC for review by January 1, 1981. ... Following NRC approval of the guidelines licensees should revise their emergency procedures by December 31, 1981, or prior to receiving an operating license, whichever is later.

Comment

The BWR Owners' Group believes they have fully complied with the requirements of Item I.C.1 to perform revised analyses and prepare emergency procedure guidelines through the submittal of the following documents:

a. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," NEDO-24708, August 1979

b. Section 3.2.1 (Revised) of NEDO-24708, "Analysis of Loss of Feedwater Events," transmitted by R. H. Buchholz letter to D. F. Ross dated March 31, 1980

c. BWR Emergency Procedure Guidelines, Revision 0 (Prepublication form), transmitted by R. H. Buchholz letter to D. G. Eisenhut dated June 30, 1980

d. Section 3.2.2 of NEDO-24708, "Other Operational Transients," transmitted by R. H. Buchholz letter to D. G. Eisenhut dated August 22, 1980

e. Section 3.5.2.1 (Revised) of NEDO-24708, "Analysis to Demonstrate Adequate Core Cooling" and Section 3.5.2.4 of NEDO-24708, "Justification of Analysis Methods," transmitted by R. H. Buchholz letter to D. G. Eisenhut dated September 16, 1980

In a seminar held with the NRC Staff in early August 1980 to review the Emergency Procedure Guidelines, the Staff indicated that, except for minor technical justification of several items and the details associated with implementing the guidelines, they were satisfied that the material submitted met the requirements of this task. The BWR Owners' Group requests that Reference 1 be modified to reflect the satisfactory completion of these tasks for the BWR.

Requirement II.B.1

Reactor Coolant System Vents

Clarification A.3

The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered is to specify a volume of noncondensible gas to be vented and in a specific venting time (e.g., a vent capable of venting a gas volume of one half of the RCS in 1 hour). For con ments particularly vunerable to failure from large hydrogen releases over a sho period of time, the capability of venting outside containment (see Item 9 below) must be considered. Other criteria and engineering approaches should be considered if desired.

Clarification A.9

Since the generation of large quantities of noncondensible gases could be associated with substantial core damage, venting to atmosphere presents difficult problems because of the associated release of radioactivity. The implications of venting either inside or outside containment must be studied. This is a new requirement, and a schedule change has been made to accommodate this study. Within containment, those areas which provide good mixing with containment air are preferred.

In addition, areas which provide for maximum cooling of the vented gases are preferred. Therefore, the selection of the location for venting should take advantage of existing ventilation and heat removal systems.

Comments

The BWR Owners' Group provided information in Reference 5 describing the design features of the BWR which satisfy the requirement for reactor coolant system vents as defined in NUREG 0578. Reference 5 identifies the Safety Relief Valves (S/RV), Reactor Head Vent line, Reactor Core Isolation Cooling (RCIC) system and High Pressure Coolant Injection (HPCI) system as all providing venting paths. In addition, the BWR is designed to operate with gases in the reactor vessel head and the presence of noncondensibles in this area does not interfere with the BWR's natural circulation capability.

Clarifications A.3 and A.9 as drafted expand the scope of this requirement and suggest that the BWR Owners' Group evaluate the capability and consequences of venting the BWR outside the containment. At the regional meeting held in Las Vegas on September 24, 1980, the staff indicated verbally that both 1) venting the RCS directly to the atmosphere and 2) venting the containment to the atmosphere should be addressed as part of Task II.B.1. The BWR Owners' Group believes the additional evaluations being requested are beyond the scope necessary to accomplish the stated purpose of the recommendation which is to "vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation."

Given the postulated condition that significant noncondensibles are released or generated following a degraded loss of coolant accident (LOCA), several potential paths already exist which may lead to a release of the noncondensibles from the vessel to the containment in addition to those paths which may be manually actuated (e.g., S/RVs). If the break which is the source of the LOCA is located on a pipe which is attached to the reactor vessel above the water level, noncondensibles will also be vented to the containment through the break. Thus, the provision of manually actuated reactor coolant system vents does not create a unique venting path.

If hydrogen is present in the noncondensibles which are vented, operator procedures are already in place to insure containment integrity is maintained for hydrogen concentration levels up to the maximum calculated using 10CFR50.46. This is true no matter which route the noncondensibles take in reaching the containment - through the S/RVs or via the break. Therefore, demonstrating that noncondensibles can be vented through S/RVs does not also require that venting to the atmosphere be evaluated.

Any requirement to evaluate hydrogen control capability for hydrogen concentration levels in excess of those calculated using 10CFR50.46 should be addressed through the rulemaking proceeding for degraded core conditions.

Requirement II.B.2

Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which May Be Used in Post Accident Operations

Clarification 4

Any plant modifications (e.g., additional shielding, new equipment, moving equipment) which are needed to protect safety equipment outside containment must be completed by January 1, 1981.

Comment

To be consistent with the schedule specified under IMPLEMENTATION DATE, the January 1, 1981 date in Clarification 4 should be changed to January 1, 1982.

Requirement II.B.3

Post Accident Sampling Capability

Clarification 3a

Provisions shall be made to permit post-accident containment atmosphere sampling under both positive and negative pressure.

Comment

Extensive analyses have been made which confirm that post-accident containment atmosphere pressure in plants with Mark I, II, and III containments 1) remains positive throughout the course of the design basis events, or 2) is negative for only a very short period during an event. This containment pressure response is assured for the Mark I, II, and III containments provided with vacuum breakers. Accordingly, it is the conclusion of both General Electric and the BWR Owners' Group that the requirement to be able to sample the containment atmosphere under negative pressure conditions is not necessary for BWR plants. It is recommended that this requirement be modified for BWRs such that provisions need not be provided to permit sampling under negative pressure conditions.

Requirement II.E.4.2

Containment Isolation Dependability

Clarification 5

Ganged resetting of containment isolation valves is not acceptable. Resetting of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, if electrical independence and other single failure criteria are met.

Comment

It is felt that the above requirement is too restrictive. It eliminates the existing capability to gang reset valves in more than one line when single failure criteria are met (i.e., where two distinct and independent operator actions are required to open one or more lines). This claifirication should be revised to allow this ganged reset capability under the conditions specified above. Otherwise, many plants would require extensive and unnecessary modifications.

Requirement II.F.1(3)

Containment High Range Radiation Monitor

Clarification 5

For BWR Mark III containments, two such monitoring systems should be inside both the primary containment and secondary containment.

Comment

Rewording of the clarification is recommended to more specifically identify exactly what is desired for the Mark III containment and what the objective of this clarification is for this limited application.

Requirement II.F.1(5)

Containment Water Level Monitor

In the NRC preliminary clarification letter, it was stated that the containment water level monitors should be installed on all operating reactors by January 1, 1981.

Comment

The implementation of this requirement will require plant outages of 3 to 4 weeks for several plants with Mark I containments due to the need to drain the torus for the installation of water level taps. These would be unscheduled outages and would adversely impact plant availability if the change must be implemented by January 1, 1981. It is recommended that the implementation date for this requirement be changed to the first scheduled outage following January 1, 1981.

Requirement II.F.1(6)

Containment Hydrogen Monitor

Clarification 3

Containment hydrogen concentrations shall be measurable over the range from 0 to 10 volume percent, with a measurement accuracy within ± 1% of the monitored range (i.e., ± 0.1 volume percent hydrogen for a 10 volume percent range).

Comment

As indicated in the comments offered by various BWR Owners' Group representatives during the NRC regional meetings, a hydrogen monitor with the measurement accuracy specified in the clarification is beyond the state of the art. No hydrogen monitoring system with the stated measurement accuracy is commercially available. Additionally, it was noted that potential operator actions based upon containment hydrogen concentrations do not dictate the need for a measurement accuracy as narrow as ± 0.1 percent. A realistic alternative to the value in the preliminary clarification such as a measurement accuracy of ± 0.7 percent hydrogen (by volume) is recommended. This measurement accuracy is compatible with commercially available hydrogen monitoring systems and would supply sufficient information for operator actions.

With respect to the specified range over which containment hydrogen shall be measured, a measurement range of 0 to 10 volume percent is inconsistant with that required by Regulatory Guide 1.97 (Revision 2). Regulatory Guide 1.97 (Revision 2) specifies 0 to 30 volume percent as the desired range of measurement. Either the clarification to II.F.1(6) or Regulatory Guide 1.97 (Revision 2) should be changed such that consistancy between the requirements is achieved.

Requirement II.F.2

Instrumentation for Detection of Inadequate Core Cooling

Clarification 6

The indications must cover the full range from normal operation to complete core uncovery. For example, water level instrumentation may be chosen to provide advanced warning of two phase level drop to the top of the core and could be supplemented by other indicators such as in-core and core exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of inadequate core cooling and to infer the extent of core uncovery. Alternatively, full range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

Comments

Clarification number 6 and presentations by the staff during the Las Vegas regional meeting on September 24, 1980 indicate that core exit thermocouples should be considered for inclusion in the BWR in addition to the present water level instrumentation. The incorporation of core exit thermocouples into the BWR design has already been considered in the development of Regulatory Guide 1.97 (Revision 2). Substantial progress has already been made during the discussions of Regulatory Guide 1.97 (Revision 2) with respect to reviewing the adequacy of the BWR's current water level instrumentation and its capability to monitor for inadequate core cooling. The need for supplementing the existing water level instrumentation with core exit thermocouples has been discussed at length with the Staff and the ACRS. The ACRS had reached the conclusion that it "is unconvinced that these thermocouples will be of significant use" in BWRs (Reference 2).

The considerations of core exit thermocouples for the BWR under the scope of task II.F.2 would be redundant to the resources already expended in the review of Regulatory Guide 1.97 (Revision 2). The BWR Owners' Group recommends that the review of Regulatory Guide 1.97 (Revision 2) be brought to completion and that it be issued. Any further need to evaluate core exit thermocouples for BWRs should be pursued only as it relates to future revisions of Regulatory Guide 1.97. Elimination of a redundant requirement under II.F.2 will allow the utilities with BWRs to apply their limited resources towards implementation of other higher priority tasks of NUREG-0660.

R[illegible]irement II.K.3.14

Isolation of Isolation Condensers On High Radiation

Position

Isolation condensers have radiation monitors on their vents. These monitors provide alarms in the control room but do not isolate the isolation condenser. The isolation condensers are currently isolated on a high radiation signal in the steam line leading to the isolation condensers. The design should be modified such that the isolation condensers are automatically isolated upon receipt of a high radiaton signal at the vent rather than at the steam line. The purpose of the change is to increase the availability of the isolation condensers as heat sinks.

Comment

The BWR Owners' Group agrees with the NRC's goal of increasing the availability of the isolation condensers under post-accident conditions. The isolation condensers are passive systems and are well suited for long term core cooling. The concern is that if this modification is not properly designed, it will have an adverse effect on the availability of the isolation condenser to accomplish its intended function. To implement this requirement properly requires more than simply rewiring the existing vent line radiation monitor alarm into the isolation circuit. Current engineering evaluations indicate the existing monitor may alarm even if there were no tube rupture simply because of the high radiation levels expected in the area of the monitors during post-accident system operation. Furthermore, it is impractical to shield the present detectors sufficiently to ensure that their alarming signifies a tube rupture.

It is the BWR Owners' Group conclusion that a modification to this system which achieves the desired results cannot be completed by January 1, 1981. Since the design of the required radiation monitoring system has not yet been finalized, and thus equipment procure[illegible]nt lead times are unknown, it is proposed that a design for this modification be submitted by January 1, 1981, in addition to a proposed implementation schedule that is based upon realistic vendor delivery dates.

Requirement II.K.3.24

Confirm Adequacy of Space Cooling for HPCI and RCIC Systems

Position

Long-term operation of the RCIC and HPCI systems may require space cooling to maintain the pump room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating current power. The RCIC and HPCI systems should be designed to withstand a complete loss of alternating current power to their support systems, including coolers, for at least two hours.

Comment

Although the Staff did not choose to clarify this item in Reference 1, it is requested that the NRC attempt to clarify it in their final clarification letter. As it stands now, this position does not account for the existing redundance and essentially creates a new design basis. The statement that, "The RCIC and HPCI systems should be designed to withstand a complete loss of alternating current power to their support systems, including coolers, for at least two hours" gives plicit consent to acceptance of the new design basis. For plants where space cooling for RHR, HPCS, RCIC and LPCS is on emergency power and emergency cooling water, loss of space cooling implies loss of a safety division and that division's injection system. Since redundant RCIC/HPCI systems are on separate safety divisions and all space cooling is assumed lost, then in effect, the Staff is requesting an analysis for the loss of two safety divisions (complete loss of alternating current power to the.r support systems, including coolers). Loss of two safety divisions does not give the appropriate credit for the designed redundance of these systems.

Requirement II.K.3.30

Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K

Clarification

Reference 1 specifies that the following task be completed by October 1, 1980:

> Based on the above background and clarification, a detailed outline of your proposed program to address this issue should be submitted. In particular, this submittal should identify (1) which areas of your models, if any, you intend to upgrade, (2) which areas you intend to address by further justification of acceptability, (3) test data to be used as part of the overall verification/upgrade effort, and (4) your estimated schedule for performing the necessary work and submitting this information for staff review and approval.

Comment

The data of October 1, 1980 proposed for the submittal of a program outline is not achievable. It is recommended that a more reasonable data for completion of this task would be 60 days after issuance of the final clarification letter.

Requirement III.A.1.2

Upgrade Emergency Support Facilities

The POSITION and CLARIFICATION statements for this requirement, in addition to the draft requirements of NUREG-0696, "Functional Criteria for Emergency Response Facilities", provide the detailed design and functional criteria for the on-site Technical Support Center and the near-site Emergency Operations Facility.

Reference 1 specifies that the complete facilities' design description be submitted by January 1, 1981, and that the facilities be operational no later than April 1, 1982.

Comment

The BWR Owners' Group considers that the schedule proposed by the NRC in Reference 1 is unrealistic. To meet that schedule, a utility would have to make commitments based on the assumption that the NRC will not alter the criteria again or that the criteria were firm enough such that detailed plans could be finalized. There is no basis at present for that type of confidence. In developing the schedule specified in Reference 1, it appears that the NRC Staff has not comprehensively studied the tasks required to implement the criteria of NUREG-0696 and has not factored in the appropriate amount of time to fully complete the tasks. We believe that plant licens s should have been contacted about the proposed schedule prior to its promulgation and after the criteria had been firmed up so that good data could have been obtained. Instead, we understand the Staff polled individuals from industry organizations or equipment vendors who either did not realize the large scale of the functional requirements or who were misunderstood by the Staff who used the information to create the schedule. In particular, requirements for seismic qualifications and criteria for reliability goals appear to be arbitrarily established, without regard to the safety related benefits to be obtained. Insufficient time is allowed in the implementation schedule to complete the development and testing work which would be required to achieve these requirements, and no basis is identified which would demonstrate the validity of these requirements in the light of the time delay in system implementation which accrues from such criteria. Additional detailed comments on facility and equipment design criteria have been provided to you in several submittals by individual utilities (e.g., Reference 4).

We believe it is important that NRC guidance be developed from a comprehensive evaluation of not only what is desirable, but also what is attainable and actually needed. The NRC Staff must acknowledge that there exists finite technical, time, and financial resources available to licensees; and that the deployment of these resources on criteria that periodically change could ultimately have a deleterious effect on the public health and safety. At the NRC's regional meeting on September 22, 1980, Mr. D. G. Eisenhut stated that it was not the NRC's intent to penalize those licensees who took the initiative to meet the NRC's deadlines despite the lack of criteria approved by NRC. Based on Mr. Eisenhut's commitment to discuss site specific applications of the new criteria, we suggest that this option be specified within the clarification to be provided in the revision of Reference 1.

GENERAL COMMENTS

Reference to Regulatory Guide 1.97 (Revision 2) and to ANSI 3.2 (Draft 3)

Throughout the preliminary clarification letter references were made to Regulatory Guide 1.97 (Revision 2) and ANSI 3.2 (Draft3) even though the final versions of these documents have not been formally issued. It is suggested that the specific applicable requirements from these documents be detailed in place of general references to the documents.

Proposed Schedule Changes

The relaxation of implementation dates as indicated in the preliminary clarification letter is needed and welcomed. However, there still remains an estimated 40 items for which the staff has requested information be supplied between now and January 1, 1981. This represents a significant amount of work for utility and vendor organizations to complete in a short time period. We also believe that the staff will have difficulty in reviewing the extensive documentation that is being requested during the first months following January 1, 1981. Thus, it is requested that the NRC reevaluate their information and documentation requests, and further revise the due dates to achieve a more realistic distribution.

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT □ 6201 S Street, Box 15830, Sacramento, California 95813

October 9, 1980

Director of Nuclear Reactor Regulation
Attention: Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket 50-312
Rancho Seco Nuclear Generating Stati
Unit 1
TMI Action Plan Requirements

Dear Mr. Eisenhut:

The Sacramento Municipal Utility District has reviewed your letter of Sept 5, 1980, clarifying requirements and providing a schedule for the implemen of TMI Action Plan items. We have some serious concerns regarding these r ments and wish to provide the following comments for your consideration be this clarification is finalized.

It appears that the requirements have not been coordinated or integrated w each other. Each task is addressed individually and its effect on other t or systems does not appear to have been considered. This presents serious duling problems since in many instances the design effort on other tasks m completed before a given task can be implemented. For example, many of th requirements result in a modification, adding instruments and controls in control room. This will probably require a significant redesign of the en control room with the addition of a number of Class I panels. This activi cannot be accomplished until a majority of the TMI modification work has b engineered.

Many of these requirements significantly increase the demand on Class I po supplies. Our studies to date indicate that we will have to add two addit diesel powered generators to our emergency power supply system. Our prese best estimate for delivery of these diesel generators is twenty months aft issuance of a purchase order. Since the detailed requirements for these u have not been determined at this time, these units will not be available f installation in less than 24 to 30 months from this date.

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The requirements result in a large addition of Class I electrical cables. The magnitude of the requirements has forced us to build an additional Class I building adjacent to our existing auxiliary building. Many of the cables required for the TMI modifications and practically all of the new electrical panels will be located in this building. Design and construction of this building will be expedited; however, construction is not scheduled to start until the end of this year, and we cannot have it completed until approximately December 1, 1981. This, I should mention, is on a double shift basis.

The District fully endorses modifications which will lead to significant improvements in plant safety. We feel that the schedule requirements in your letter will lead to less than the desired results, and feel it to be highly desirable to take a more organized and thorough design approach to the requirements. We urge that you consider some flexibility in your final clarification.

Following are our specific comments on the individual requirements, organized by action plan numbers, corresponding to those in your letter of September 5, 1980.

ACTION PLAN NO.	REMARKS
II.B.1	Present estimates for the delivery of Class I valves to used in the reactor coolant vent system are in excess o 1 year. It is unrealistic to require installation of these vents by 1-1-82. Since Rancho Seco Unit 1 is the District's only thermal generating unit we do not have a large supply of spare parts or the ability to draw fro purchases for other plants under construction as a sourc for such parts. We urge consideration be given to a schedule relaxation, which would allow these valves to b installed during a regular extended refueling outage, following valve delivery. This is presently scheduled f the fall of 1982. These comments on hardware availabili and implementation schedule apply in general to all hard ware modifications required by the action plan.
II.B.4	The training program for mitigating core damage will hav to be modified as additional equipment and systems are installed. The requirement should clearly specify that the training program, and not the training of each indiv dual operator, be completed by the required date.

ACTION PLAN NO.	REMARKS
II.D.1	The required dates for completion of valve testing appeared to be unrealistic. The District is participating in the EPRI testing program and the new requirement for block valve testing will have an as yet undete mined effect on the overall schedule.
II.E.1	The District is participating with Arkansas Power and Florida Power in a joint effort for the design of a new Class I initiate and control system for the auxiliary feedwater system. This upgrade was discussed with your staff at a meeting on September 4, 1980. The result of this upgrade will be an auxiliary feedwater system whic can be classified as an engineered safety feature, and include features to insure natural circulation, prevent over cooling, prevent over filling, and result in a sig nificant safety improvement at Rancho Seco. As discussed with your staff, hardware cannot be procured for this improvement until the first quarter of 1982; and w would not anticipate installation until later that year This schedule is consistent with the completion of the new building which will house most of the equipment. A detailed description of this system will be provided your staff by the middle of this month for conceptual approval.
II.E.4.1	This item is an example of a requirement that for some reason coincides with only one other requirement date. A modification to the containment penetrations is a major undertaking and the District urges that consideration be given for an implementation data coinciding with an extended outage.
II.E.4.2	This is a new requirement and the District feels that any modifications to the containment isolation signals and setpoints will require significant detailed analys We feel the July 1, 1981 date for implementation of th modifications to be totally unrealistic.
II.F.1	The required implementation date is again unique in th list of dates in your requirements and the District wo prefer that these modifications be accomplished during an extended outage. The District is unable to comment during the short period of time allowed for these comm and provide realistic dates for the installation of th equipment. Our best information shows that much of th instrumentation is unavailable at this time. The Dist

ACTION PLAN NO.	REMARKS
II.F.1 (cont'd.)	requests further time to determine what instrumentation is available and will provide meaningful input when this information is available.
II.F.2	The District has concluded, as detailed in our lette of August 28, 1980, that additional instrumentation the detection of inadequate core cooling is not necessary. We feel that existing instrumentation adequat satisfies the intended purpose of detecting and resp ing to inadequate core cooling, and we object to thi requirement in its entirety.
II.K.3	Item 30 of this plan requests a schedule and outline a small break analysis in compliance with 10 CFR 50. by October 1, 1980. The District is currently devel such a program in conjunction with other members of B&W owner's group and should be prepared to submit t program by approximately November 1, 1980.
III.D.3.3	At this time, the District does not believe the requ instrumentation for iodine measurement is available therefore, is unable to provide a date for implement

We would like to emphasize the fact that we are willing to work with you to develop dates for the implementation of requirements which will allow signifi cant improvements to be made to the safety of Rancho Seco, Unit No. 1. If we can provide any additional information, please advise.

Sincerely yours,

John J. Mattimoe

John J. Mattimoe
Assistant General Manager
and Chief Engineer

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