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Docket Nos. 50-206
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Gentlemen:

SUBJECT: EMERGENCY PLANNING QUESTIONS AND POSITIONS
(San Onofre Nuclear Generating Station, Units 1, 2 and 3)

As a result of our review of Emergency Planning at the San Onofre Nuclear Generating Station, we find that we need the information listed in the Enclosure. Almost all of these questions were transmitted to your staff at the September 27, 1979 meeting in San Clemente, California. We request that you submit your response to the Enclosure within five weeks of receipt of this letter. Please contact us if you have any questions about the information requested.

Sincerely,

Original signed by
Robert L. Baer

Robert L. Baer, Chief
Light Water Reactors Branch No. 2
Division of Project Management

Enclosure:
Request for Additional
Information

ccs w/enclosure:
See next pages

APP
3

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SAN ONOFRE EMERGENCY PLAN QUESTIONS AND POSITIONS

435.1 Describe the location and role of the onsite technical support center. See Appendix A (of this Enclosure).

435.2 Describe the location of the onsite operational support center. See Appendix B.

435.3 Table 5-10 on page 5-20 of the Unit 1 Emergency Plan needs to be updated to that found in page 5-22 of the Unit 2 & 3 plan.

435.4 Identify the onsite capability and resource to properly access and categorize accidents. Specifically address the instrumentation for detection of inadequate core cooling (See Appendix C) and radiation monitoring capability (see Appendix D). The radiation monitoring equipment identified on page 7-14 of the Unit 1 plan will require upgrading. Table 7.4 of the Unit 2 & 3 plan does not identify the operational effluent monitoring system. (See also item 435.7)

435.5 What provisions have been made for dissemination of educational information to the public within the plan exposure Emergency Planning Zone regarding the warning procedures to be used in the event of a serious accident?

435.6 Describe the resources that will be used if necessary to provide early warning and/or protective action instructions to the populace within the Emergency Planning Zone associated with the plume exposure pathway within 15 minutes following notification from the facility operator to the offsite authorities.

435.7 Identify the onsite capability and resources to provide valid and continuing assessment through the course of an accident including (See also item 4):

a. Post accident sampling (See Appendix E & G)

b. In-plant iodine instrumentation (See Appendix F & H)

c. Plots showing the containment radiation monitor vs. time following an accident for incidents involving 100% release of coolant activity, 100% release of gap activity, 1% release of fuel inventory, and 10% release of fuel inventory.

435.8 Page 7-20 of the Unit 1 Plan should be revised to reflect the upgraded environmental monitoring program specified in Table 7.7, page 7-26 of the Unit 2 & 3 Plan.

435.9 What is the location of the contractor that will provide the offsite radiological analysis specified in Table 7.7? This information is necessary in order to assess the timeliness of radioanalysis support in times of emergency conditions.

435.10 What agencies will allocate resources i.e., manpower and equipment to perform real-time radiological field assessments of accidental radiological releases? Describe the methods and equipment to be employed in determining the magnitude and locations of any radiological hazards following radioactive releases.

435.11 Are the agreement letters in Appendix A of the Emergency Plan still valid? What provisions are there for insuring their continued validity?

435.12 Describe how the public will be notified of evacuation or other protective measures within the Emergency Planning Zone associated with the plume exposure pathway. (See also item 435.6).

435.13 The San Onofre emergency plan must provide in addition to the drills and exercises identified a Regulatory Guide 1.101, a joint exercise involving Federal, State, and local response organizations.

The scope of such an exercise should test as much of the emergency plans as is reasonably achievable without involving full public participation. Definitive performance criteria should be established for all levels of participation to assure an objective evaluation. This joint test exercise will be scheduled about once every five years.

435.14 Chapter 4 should incorporate the emergency classes, the initiating conditions and the immediate actions identified in Appendix I. The example initiating conditions for each class should be supplemented, where possible, by the specific plant instrumentation readings which will initiate the emergency class. Table 4.1 is an excellent means of portraying the immediate actions being taken by <u>specific</u> utility personnel. Therefore, it is recommended that this table be kept and expanded as necessary to incorporate the guidance provided in Appendix I.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Onsite Technical Support Center (Section 2.2.2.b)

1. INTRODUCTION

Each applicant for a construction permit is required by 10 CFR 50.34(a) to include in its PSAR a discussion of preliminary plans for coping with emergencies. Each applicant for an operating license is required by paragraph 50.34(b) to include plans for coping with emergencies in its FSAR. Appendix E to 10 CFR Part 50 establishes minimum requirements for emergency plans. Regulatory Guide 1.101 provides more complete guidance to be used in developing the emergency plans required in FSARs for nuclear power plants. These plans are described in the PSAR and are submitted as a part of the FSAR. They do not consistently cover the role of technical and management personnel during an emergency. Similarly, there are no detailed regulatory requirements concerning the need for technical information on plant status and operation outside of the control room during off-normal events. The capability to transmit and record vital plant data in real-time is also not a current requirement, nor is it required that as-built plant drawings and updated records be available to support emergency activities.

The purpose of this recommendation is to establish a center outside of the control room that acts in support of the command and control function and to improve plant status and diagnostic information at this location for use by technical and management personnel in support of reactor command and control functions.

2. DISCUSSION

The recommendations given above for the role of the shift supervisor, the addition of a shift technical advisor, and the limitation of control room access are to be complemented by this recommendation to require the establishment of an onsite technical support center. The activities of plant engineering and management personnel are an important part of the overall station response to an accident and must be properly defined and logistically supported. These people provide the in-depth technical support of control room activities and typically are responsible for the implementation of emergency procedures.

During the first 2 days following the accident at TMI-2, it was difficult for senior government officials to establish contact with senior plant management. It is anticipated that the onsite technical support center will serve as the focal point for such communication in the future.

There is also an indication from the events at TMI-2 that implementation of emergency plans by personnel in the control room acted to congest and confuse the reactor operations control activities. The technical support center would provide a place, in close communication with the control room so as to have sufficient knowledge of current and projected plant status, for more orderly implementation of emergency procedures.

Review of the TMI-2 accident also shows a lack of reliable technical data, information, and records on which to base accident recovery decisions. Knowledgeable nuclear engineers were unable to understand the details of plant conditions or plant design so as to better advise the operators of appropriate actions for accident recovery.

On many occasions subsequent to the March 28 accident, as-built drawings reflecting the actual configuration of critical portions of the plant were either not available or contained erroneous information. This situation contributed to delays in accident recovery.

Over the long term, it will probably be useful to provide plant status monitoring and recording equipment in the onsite technical support center. The Task Force recommends that requirements in this regard be developed in conjunction with requirements concerning the kind and form of information to be transmitted to the NRC.

3. POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

A complete set of as-built drawings and other records, as described in ANSI N45.2.9-1974, shall be properly stored and filed at the site and accessible to the technical support center under emergency conditions. These documents shall include, but not be limited to, general arrangement drawings, P&IDs, piping system isometrics, electrical schematics, and photographs of components installed without layout specifications (e.g., field-run piping and instrument tubing).

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Onsite Operational Support Center (Section 2.2.2.c)

1. INTRODUCTION

Each applicant for a construction permit is required by 10 CFR 50.34(a) to include in its preliminary safety analysis report a discussion of preliminary plans for coping with emergencies. Each applicant for an operating license is required by paragraph 50.34(b) to include plans for coping with emergencies in its final safety analysis report. Appendix E to 10 CFR Part 50 establishes minimum requirements for emergency plans. Regulatory Guide 1.101 provides more complete guidance to be used in developing the emergency plans required in FSARs for nuclear power plants. These plans do not consistently cover the role and logistical support for operations support personnel during an emergency.

The purpose of this recommendation is to establish a primary operational support area, to be designated as the onsite operational support center, for shift personnel to be in direct communication with the control room and other operations managers for assignment to duties in support of emergency operations.

2. DISCUSSION

During the TMI-2 accident, operational support personnel (e.g., auxiliary operators not assigned to control room, health physics personnel, and technicians) reported to the control room. This contributed to the congestion and confusion in the control room. Although these personnel are required for operations outside of the control room and perhaps a few in the control room, there is a need to restrict their access to only those specifically requested by the shift supervisor to be present in the control room. Thus, there is a need to establish an area in which shift personnel report for further instructions from the operations staff.

3. POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs (Section 2.1.3.b)

1. INTRODUCTION

General Design Criterion 13, "Instrumentation and Control," of Appendix A to 10 CFR 50, requires instrumentation to monitor variables "... for accident conditions as appropriate to assure adequate safety." In the past, GDC 13 was not interpreted to require instrumentation to directly monitor water level in the reactor vessel or the adequacy of core cooling. The instrumentation available on some operating reactors that could indicate inadequate core cooling includes core exit thermocouples, cold leg and hot leg resistance temperature detectors (RTDs), in-core neutron detectors, ex-core neutron detectors, and reactor coolant pump current. Generally, such systems were included in the reactor design to perform functions other than monitoring of core cooling or indication of vessel water level.

During the TMI-2 accident, a condition of low water level in the reactor vessel and inadequate core cooling existed and was not recognized for a long period of time. This problem was the result of a combination of factors including an insufficient range of existing instrumentation, inadequate emergency procedures, inadequate operator training, unfavorable instrument location (scattered information), and perhaps insufficient instrumentation.

The purpose of this recommendation is to provide the reactor operator with instrumentation, procedures, and training necessary to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling.

2. DISCUSSION

With the hindsight of TMI-2, it appears that the as-designed and field-modified instrumentation at Three Mile Island Unit 2 provided sufficient information to indicate reduced reactor vessel coolant level, core voiding, and deteriorated core thermal conditions.

To preclude the failure to recognize such conditions in the future, it is appropriate to address the problem in two stages. The first is based on the detection of reduced coolant level or the existence of core voiding with the existing plant instrumentation. This would include wide range core exit thermocouples, cold leg and hot leg RTDs, coolant inventory control, in-core and ex-core detectors, vessel level (BWR), reactor coolant pump current, and other indications of coolant conditions, including coolant saturation meters (PWR). The second stage is to study and develop system modifications that would not require major structural changes to the plant and that could be implemented in a relatively rapid manner to provide more direct indication than that available with present instrumentation. These changes include PWR vessel level detectors.

A number of ideas have been discussed for the second stage by the NRC Division of Reactor Safety Research, the ACRS, and the reactor vendors. Some of the possibilities include pressure differential cells, conductivity probes, heated thermocouples, ultrasonic sounding, as well as gamma and neutron void detectors. However, we conclude that detailed engineering evaluation is required before design requirements for a direct level measurement system can be specified.

3. POSITION

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Increased Range of Radiation Monitors (Section 2.1.8.b)

1. INTRODUCTION

Monitors for radioactive effluents are designed to detect and measure releases associated with normal reactor operations and anticipated operational occurrences. Such monitors are required to operate in radioactivity concentrations approaching the minimum concentrations detectable with "state-of-the-art" sample collection and detection methods. These monitors comply with the criteria of Regulatory Guide 1.21 with respect to releases from normal operations and anticipated operational occurrences.

Radioactive gaseous effluent monitors designed to operate under conditions of normal operation and anticipated operational occurrences do not have sufficient dynamic range to function under release conditions associated with certain types of accidents. General Design Criterion 64 of Appendix A to 10 CFR Part 50 requires that effluent discharge paths be monitored for radioactivity that may be released from postulated accidents. The gaseous effluent monitoring system for TMI was evaluated during the licensing review and was found to be adequate for calculated releases from postulated accidents; however, the TMI experience gives rise to a new interpretation of postulated accidents and their associated releases.

The radiation level inside containment is a parameter closely related to the potential for release of radioactive materials in plant effluents. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," requires (for plants whose submittals for construction permit applications were docketed after September 30, 1977) the capability for measuring in-containment radiation levels up to 10^8 rad/hr.

2. DISCUSSION

At TMI-2, the noble gas section of the gaseous radioactive effluent monitor serving the plant vent was designed to measure effluent concentrations up to 10^{-2} μCi/cc (Xe-133). During the initial phases of the accident, noble gas radioactive effluent readings were off scale, with estimates of actual release concentrations calculated to be on the order of 10^{-1} μCi/cc to 1 μCi/cc.

Similarly, a section of the TMI plant vent gaseous radioactive effluent monitor designed to detect and measure radioiodine releases, while remaining on scale, gave an erroneous indication of high radioiodine content in releases from the vent during the initial phases of the accident. The indication was caused by concentration of short-lived noble gases in the charcoal cartridge, with the presence of the noble gases being read and erroneously interpreted as radioiodine by the monitor readout system.

A similar condition existed in the section of the plant vent monitor designed to detect and measure the presence of particulate radioactive material in plant gaseous effluents. In this case, the presence of noble gases in the gas stream passing through the monitor's particulate filter was sufficient to cause the particulate section of the monitor to read off scale and erroneously indicate that large quantities of particulates were being released from the plant vent.

The problem is considered to be generic. A recent survey of existing gaseous effluent monitoring capabilities of operating plants shows that less than 20 percent of operating plants have monitors that would have stayed on scale under the conditions of the TMI accident. It can also be shown, however, that the potential releases from postulated accidents may be several orders of magnitude higher than was encountered at TMI. Under such circumstances, none of the effluent monitors now in service at any operating plant would remain on scale.

A gaseous radiological effluent monitor that does not provide on-scale readings under accident conditions provides only lower-bound information on effluent releases to the environment. A requirement for effluent monitors to have an operating range sufficient to permit on-scale readings under accident conditions is needed to provide meaningful release information for off-site emergency actions.

Three components of gaseous effluents are usually monitored. These are (a) noble gases (for gross activity relative to xenon-133 calibration); (b) radioiodines (usually sampled by collection on charcoal and detected and measured either on the basis of gross gamma activity, which assumes all activity to be iodine-131, or on the basis of a single-channel sodium iodide gamma spectrometer centered on the 0.364 Mev peak of I-131); and (c) particulates (for gross activity collected on a paper or fiber filter relative to a calibration source such as cesium-137).

Under normal operating conditions, a three-component effluent monitoring system is capable of functioning in accordance with design. Readout, under normal operating conditions, provides the plant operator with a reasonably accurate continuous measurement of the actual instantaneous release concentration of noble gases. However, the measurements of radioiodine over a given time period are based on the accumulation of airborne particulates or radioiodine over a given time period in the filter or adsorption media. It is necessary for the plant operator to separately calculate the effluent concentration of interest on the basis of the time rate-of-change of the monitor readout. (Note: Recent improvements involving the use of microprocessors have made it possible to obtain instantaneous effluent concentrations from integrating-type measurement data by continuous calculation of the time rate-of-change using a built-in computing system.)

The NRC staff recently conducted a survey of installed noble gas effluent monitors at 56 of the 69 operating nuclear units. The survey indicates that nine reactors have effluent monitors whose range exceeds 100 Ci/sec. These monitors would probably have stayed on scale during most of the TMI-2 accident. The remaining reactors have monitors that would have been off scale for various segments of the early days of the accident. Thirty-seven of the 56 reactors have monitors with an upper range that is below 10 Ci/sec. Most of the reactors

(59 out of 66) have monitors with an upper range that exceeds that of the TMI-2 station vent monitor, which was off scale at about 0.5 Ci/sec.

Based on data submitted by plant operators, the installed capability exists for monitoring noble gas releases up to a concentration of approximately 1×10^3 μCi/cc, which is a factor of 10^5 higher than the maximum range of the instrumentation in use of TMI.

The Task Force notes the recent publication of ANSI N320-1978, "Performance Specification for Reactor Emergency Radiological Monitoring Instrumentation," effective December 6, 1978. ANSI N320-1978 recommends an upper detection limit of 10^5 μCi/cc for noble gases released to the environs through plant stacks. The staff considers the upper detection limit of 10^5 μCi/cc for noble gases to be technically achievable.

The staff understands that technological problems exist in monitoring of particulates and radioiodines in potential plant releases. Completely satisfactory equipment apparently is not currently available on the commercial market. As previously discussed, the accident condition results in the presence of comparatively large concentrations of short-lived noble gases, which the detectors of the particulate and iodine monitor components "see" as particulates and radioiodines. The problem is further compounded by the preferential adsorption of noble gases in the charcoal cartridges. Although the noble gases are not retained for any substantial period of time, the net effect of a continuous flow of gases through the charcoal cartridge is a localized concentration of noble gases, which is "seen" by the radioiodine detector as radioiodine. Under normal operating conditions, the radioiodine detector is operated as a single-channel gamma spectrometer, focussing on the 0.364 Mev peak of I-131 and rejecting the normally encountered Xe-133 and Kr-85. Under accident conditions, however, the short-lived noble gases are present, several of which emit gamma photons near the 0.364 Mev gamma of I-131, thus being registered as I-131 on the monitor readout. In addition, accident levels of I-131 concentrated on the charcoal cartridge in close proximity to the detector can accumulate to the extent of saturating the detector.

It has been suggested that other adsorbents may be found that would preferentially concentrate the radioiodines, but not the noble gases. If this is found to be practicable, this could somewhat alleviate the radioiodine monitoring dilemma; however, the short-lived noble gases would still be present in the airstream passing through the monitor and the monitor would still give false data. At this time, there are no demonstrated techniques and no currently available equipment that will provide for the desired monitoring of radioiodines or particulates in plant gaseous effluents under accident conditions.

The Task Force concludes that sampling of plant gaseous effluents, with laboratory analysis of samples subsequent to release, is the only valid technique for monitoring accidental releases of radioiodines and particulates. In the absence of valid on-line monitoring capability for accident-level releases of radioiodines and particulates, we strongly urge that research be undertaken promptly to develop such capability.

The Task Force is working with other members of the NRC staff to urge that the NRC promptly adopt ANSI N320-1978 in its entirety, including those provisions dealing with radiation measurements in containment and other plant buildings, airborne radioactivity measurements within the plant, and airborne radioactivity measurements and radiation measurements in the environment. Implementation of the standard should take place as soon as practical for those criteria consistent with available equipment. It is further urged that research programs be established for development of instrumentation and equipment to meet the criteria that cannot be met by currently available equipment. The mechanisms suggested for implementation include adoption by reference of certain criteria in a revision to Regulatory Guide 1.97 and preparation of one or more additional Regulatory Guides to implement the remaining criteria.

At TMI-2, the radiation monitor in containment had a range capacity of 10^6 rad/hr, which was adequate to meet the conditions of the accident. In reviewing the monitoring capabilities of other plants, however, it is found that there are few operating plants with instrumentation capable of measuring levels in excess of 10 rad/hr. During the initial post-accident period at TMI, questions arose as to the validity of the instrument readout and to the operational characteristics of the instrument under the accident environment. The Task Force considers that the in-containment high-level monitoring instrumentation at TMI-2 was adequate to measure the existing radiation levels; however, it also considers that such instrumentation should consist of at least two channels, each separated physically from the other, and that the instrumentation system should be qualified to the design criteria for safety-grade instrumentation. Furthermore, the in-containment radiation monitor should be capable of measuring radiation up to 10^8 rad/hr, as currently required in Regulatory Guide 1.97. The Task Force also recommends that the instrumentation described above be required for all operating plants and for all plants now under construction.

3. POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

    a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

    b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from a minimum of 10^{-7} μCi/cc (Xe-133) to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Improved Post-Accident Sampling Capability (Section 2.1.8.a)

1. INTRODUCTION

Prompt sampling and analysis of reactor coolant and of containment atmosphere can provide information important to the efforts to assess and control the course of an accident. Chemical and radiological analysis of reactor coolant liquid and gas samples can provide substantial information regarding core damage and coolant characteristics. Analysis of containment atmosphere (air) samples can determine if there is any prospect of a hydrogen reaction in containment, as well as provide core damage information.

No definitive regulatory requirements exist for obtaining and analyzing reactor coolant samples following an accident. Standard Review Plan Section 9.3, "Process Sampling System," and Section 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," require that reactor coolant sampling provisions exist; however, no mention of accident conditions is made and, historically, this requirement has been understood to apply only to normal conditions. Standard Review Plan Section 12.5, "Health Physics Program," specifies radiological analysis requirements for liquid and gas samples under "routine" conditions, which does not include major accidents.

Standard Review Plan Section 6.2.5, "Combustible Gas Control in Containment," requires the capability to monitor containment air hydrogen levels under accident conditions. It does not, however, specifically require the capability to obtain and analyze a sample of containment air. Regulatory Guide 1.97, "Instrumentation to Follow the Course of An Accident," addresses on-line instrumentation and does not directly address the acquisition and analysis of liquid or gas samples.

2. DISCUSSION

Timely information from reactor coolant and containment air samples can be important to reactor operators for their assessment of system conditions and can influence subsequent actions to maintain the facility in a safe condition. Following an accident, significant amounts of fission products may be present in the reactor coolant and containment air, creating abnormally high radiation levels throughout the facility. These high radiation levels may delay the obtaining of information from samples because people taking and analyzing the samples would be exposed to high levels of radiation. In addition, the abnormally high background radiation, high sample radiation, and high levels of airborne contamination may render in-plant radiological spectrum analysis equipment inoperable during and after an accident.

At TMI-2, all of the above problems were encountered. The licensee was not prepared to obtain and analyze in a timely manner the reactor coolant and containment air samples under accident conditions. The acquisition of reactor coolant and containment air samples was delayed for several days while personnel radiation protection precautions were taken. Once the samples were obtained, there were significant delays in the radiological spectrum analysis of the samples. The TMI spectrum analysis equipment was inoperable because of high background radiation; consequently, the samples had to be packaged and flown to a Department of Energy (DOE) laboratory for radiological analysis.

In summary, the radiation at TMI caused by the accident delayed acquisition of information to confirm that significant core damage had occurred. Prompt acquisition and spectrum analysis of reactor coolant samples within several hours after the initial scram would have indicated that significant core damage had occurred; perhaps with such information, earlier remedial actions could have been taken. Similarly, analysis of an early containment air sample would have indicated the presence of hydrogen, significant core damage, and the possibility of a hydrogen explosion in the containment.

3. POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) quantify certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Improved In-Plant Iodine Instrumentation (Section 2.1.8.c)

1. INTRODUCTION

10 CFR Part 20 provides criteria for control of exposures of individuals to radiation in restricted areas, including airborne iodine. Since iodine concentrates in the thyroid gland, airborne concentrations must be known in order to evaluate the potential dose to the thyroid. If the airborne iodine concentration is overestimated, plant personnel may be required to perform operations functions while using respiratory equipment, which sharply limits communication capability and may diminish personnel performance during an accident. The purpose of this recommendation is to improve the accuracy of measurement of airborne iodine concentrations within nuclear power plants.

2. DISCUSSION

The concentration of iodine in atmospheric air is determined by measuring the activity of iodine adsorbed in a carbon filter through which air has been pumped. The charcoal filter is removed from the air pump and allowed to ventilate to permit the noble gases to diffuse to the atmosphere. The filter is then counted for radioactivity content and the remaining activity is ascribed to iodine. This procedure is conservative; however, it is possible for sufficient noble gas to be adsorbed in the charcoal so that the resulting iodine determination may be unduly conservative (high). This was the case at Three Mile Island. Because the iodine concentration was greatly overestimated, plant personnel performed their operations functions using respiratory equipment when such use was not necessary. Actual iodine concentrations apparently were below levels requiring such protective actions. One acceptable method to eliminate this problem is to measure the iodine by gamma energy spectrum analysis. Equipment for such measurements is commercially available.

3. POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

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Appendix G

IMPLEMENTATION OF SHORT TERM ACTIONS RECOMMENDED
BY THE LESSONS LEARNED TASK FORCE, SECTION 2.1.8.a
"IMPROVED POST-ACCIDENT SAMPLING CAPABILITY"

I. PURPOSE

The primary purpose of implementing Improved Post-Accident Sampling Capability (Item 2.1.8.a) is to improve efforts to assess and control the course of an accident by:

A. Providing information related to the extent of core damage that has occurred or may be occurring during an accident;

B. Determining the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment;

C. Providing information to determine if there is a potential for a hydrogen explosion in the containment.

The above information requires a capability to perform the following analyses:

A. Radiological and chemical analysis of pressurized and unpressurized reactor coolant liquid samples.

B. Radiological and hydrogen analysis of containment atmosphere (air) samples.

II. ACCEPTANCE CRITERIA

The following criteria will be used in the determination of the acceptability of a licensee's facility to obtain the data described above:

A. Post Accident Sampling

1. The facility shall have capability to promptly obtain (in less than 1 hour) the following samples under accident conditions:

   Pressurized and unpressurized reactor coolant

   Containment atmosphere (air)

   In the sampling process:

   a) The licensee shall have a capability to obtain representative samples. For the reactor coolant samples, sample points should be located in turbulent flow zones. For the containment atmosphere sample, the sample shall be taken in such a location as to assure a representative sample. As applicable, additional guidance is given in Regulatory Guide 1.21, paragraph C.6, and ANSI 13.1 - 1969.

   b) Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

   c) The licensee shall consider provisions for purging sample lines, for reducing plateout in sample lines, minimizing sample loss or distortion, for the prevention of blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, for the failure of isolation valves to the closed position, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

d) If changes or modifications to the existing sampling system are required in order to satisfy these criteria, the seismic design and quality group classification of sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and piping downstream of the second isolation valve can be designed to quality group D and nonseismic Category I requirements.

2. Additional samples shall be capable of being taken, as needed, in order to acquire updated information during the course of an accident.

B. Post Accident Sampling Analysis

1. The licensee's radiological spectrum analysis facility (RSAF) shall have the capability to:

a) Provide within 2 hours quantification of certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums, and non-volatile isotopes).

b) Measure the isotopes of the nuclide categories discussed above to levels corresponding to Regulatory Guide 1.3 or 1.4 for a containment air sample. Measure the isotopes discussed above up to levels corresponding to Regulatory 1.7 (with a Regulatory Guide 1.3 or 1.4 noble gas source term dissolved in the water) for a reactor coolant sample. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel exposure, should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/gm to the upper levels indicated here.

c) Restrict background levels of radiation in the RSAF from sources outside the RSAF and other sample sources inside the RSAF such that the sample analysis will provide results with an error of less than ± 100%. This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system designs which will control the presence of airborne radioactivity.

d) Maintain plant procedures which identify the analyses required, measurement techniques and provisions for reducing RSAF background levels.

2. The licensee shall have the following chemical analysis capabilities:

   a) Provide within 2 hours, quantification of hydrogen levels in the containment atmosphere.

   b) Provide within 1 hour, quantification of boron concentrations of liquids.

   c) Provide within 8 hours, quantification of chloride concentrations of liquids.

   d) The procedures for the chemical analyses listed above shall consider the presence of a source term as indicated above. Where necessary, capability for dilution or chemical separation should be provided.

   e) Measure the hydrogen concentration in the containment atmosphere in the range of 0 to 10 volume percent.

C. Post Accident Sampling and Analyses - Personnel Exposure Criteria
It should be possible to obtain and analyze a sample (or samples) while incurring a radiation dose to any individual that is as low as reasonably achievable and not in excess of 3, 7-1/2 and 18-3/4 rems to the whole body, skin or extremities, respectively. In assuring that these limits are met, the following criteria will be used by the staff:

1. For shielding calculations, Regulatory Guide 1.3 or 1.4 source terms shall be used for sample lines carrying containment air. For sample lines carrying reactor coolant, shielding calculations should be based on Regulatory Guide 1.7 and on a Regulatory Guide 1.3 or 1.4 noble gas source term in the coolant. Fifteen minutes of radioactive decay may be assumed for these calculations for the sample room and 1 hour of radioactive decay may be assumed for the RSAF.

2. Access to the sample station and to the RSAF and chemical analysis facility (CAF) shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria of II.C are met. If the existing sample station and RSAF at the licensees facility do not satisfy these criteria, then additional design features and shielding to provide access to the sample station and RSAF shall be provided.

3. Operations in the sample station, handling of highly radioactive sample from the sample station to the RSAF and CAF and handling while working with the samples in the RSAF and CAF shall be such that the radiation dose criteria of II.C are met. This shall involve the sufficient shielding of sample lines in the sample station and sufficient shielding of personnel from the samples and/or the dilution of samples for analysis in the

RSAF. If the existing sampling station or RSAF at the licensee's facility do not satisfy these criteria, then additional sampling and/or analysis design features and additional shielding shall be provided. The radioactive sample lines in the sample station, the samples themselves in the RSAF and CAF and other radioactive lines in the vicinity of the sampling station and RSAF and CAF shall be evaluated if additional shielding is necessary.

4. Plant procedures shall be provided which document sampling operations and the handling and analysis of samples and assure that plant personnel are following the approved sampling and analysis procedures which keep radiation exposure below the radiation criteria defined in II.C. As part of these procedures, high range portable survey instruments and personnel dosimeters should be provided to permit rapid assessment of high exposure and accumulated personnel exposure.

III. INFORMATION REQUIRED

The following information shall be supplied by the licensee in order to demonstrate that the acceptance criteria of Section II have been met:

The licensee shall describe the capability of his sampling system and RSAF and CAF to meet the criteria of II.A. and II.B. This description should include:

1. A description of the sampling system, including the P&ID's showing the location of sample points, and the ranges of system pressure and temperature under which samples can be obtained.

2. A description of the RSAF and CAF, an indication of the capability to analyze for the radionuclides of Item II, hydrogen, boron and chloride, including methods used in analyzing those samples. Indicate the lower and upper range of radioactivity concentrations that can be analyzed and describe how this range satisfies the criteria of Item II.

3. A description of the measures that will be instituted to ensure that background radiation in the RSAF and CAF does not affect the capability to obtain sample analysis data with an acceptably small error. The description should indicate the background sources both outside the room and inside the room, including other samples in the RSAF and CAF and any airborne contamination in the RSAF.

4. Where the existing systems do not meet the criteria given above, a description of the additions or modifications required to meet the criteria. Provide a discussion of the seismic design criteria and quality group classification of any modifications or changes, as appropriate.

5. A discussion of the methods to be used to determine that personnel involved in the obtaining of samples and the analysis of samples receive exposures that meet the criteria of II.C, including the following:

   a) A calculation, with bases and assumptions, of estimated personnel exposures during a typical sampling and analysis procedure, including exposures during access to the sample station, RSAF and CAF. Indicate the routes which personnel must take to reach the sample station and reach the RSAF and CAF from the sample station and from the control room for every person involved in the sampling and analysis process. If additional shielding, revised access, or other design changes such as an alternative on-site RSAF and/or CAF locations are necessary to keep exposure levels within the criteria, a discussion of this should be presented.

   b) An outline of the sampling and analysis procedures that will be used (details of the exact procedures need not be provided). Indicate the frequency with which these procedures are reviewed. Describe the portable instruments available for post-accident exposure survey, and the range of these instruments.

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Appendix H

Improved In-Plant Iodine Instrumentation Under Accident Conditions (2.1.8.c)

I. Requirement

The facility shall: 1) have the capability to obtain promptly liquid and air samples from throughout the plant under accident conditions, and 2) be able to accurately determine the iodine concentrations of these samples. In addition, each licensee shall provide the necessary equipment and associated training, and procedures to perform such measurements.

II. Use of Portable versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the plant buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve charcoal filters after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

For these reasons, portable instrumentation shall be available for use to detect concentrations of iodine in areas with airborne contamination.

III. Iodine Filters and Measurement Techniques

A. Charcoals impregnated with amines, such as TEDA, and silver zeolites are very efficient in adsorbing all forms of iodine. These filter media are commercially available and should be used for use in the entrainment of airborne concentrations or radioiodine. Detectors should have shielding to reduce background to levels acceptable for counting the activity collected on the charcoal.

B. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980.

1. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This shall be accomplished by using a portable or cart-mounted charcoal filter iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 264 keV of ^{131}I. A representative air sample

shall be taken and then counted for ^{131}I using the SCA. This will give an initial estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system will not saturate as a result of too much activity collected on the charcoal.

2. The licensee shall be capable to then remove the charcoal filters to a low background, low contamination area for further analysis. This "clean" area should be provided with clean filtered air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the charcoal filters. Here, the filters should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. Once purged, the presence of iodine can be detected by counting the charcoal cartridge for gross gamma using a GM detector or SCA. The licensee shall have the capability to measure accurately the iodine concentrations present on these filters.

C. The following spectrum analysis capability to perform a gamma energy spectrum analysis on charcoal filters to determine what isotopes of iodine are present. This analysis shall be performed in a low background, low contamination area.

D. The licensee shall remain up to date on the state-of-the-art methods for iodine detection. Brookhaven National Laboratories has developed an inorganic adsorber which can eliminate the noble gas retention problems associated with charcoal filters. Although not yet commercially available, this 5% silver impregnated silicon gel adsorber is better than 90% efficient for any known species of iodine under all expected temperature and humidity conditions. Its noble gas adsorption efficiency is at least 2 orders of magnitude smaller than that for iodine. This high affinity for iodine with respect to noble gases allows using this adsorber. Test results demonstrating the effectiveness in iodine detection of this adsorber are contained in NUREG/CR-0314, "An Air Sampling System for Evaluating the Thyroid Dose Commitment Due to Fission Products Released from Reactor Containment," C. Distenfeld and J. Klemish, Brookhaven National Laboratories.

E. All personnel taking airborne samples and analyzing the cartridge filters for iodine shall be adequately trained in these areas.