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June 25, 1979

U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region V
Suite 202, Walnut Creek Plaza
1990 North California Boulevard
Walnut Creek, California 94596

Attention: Mr. R. H. Engelken, Director

Docket No. 50-206
San Onofre Unit 1

References: (1) SCE (J. H. Drake) letter to NRC (R. H. Engelken) dated May 3, 1979, Docket No. 50-206

(2) SCE (J. T. Head, Jr.) letter to NRC (R. H. Engelken) dated May 23, 1979, Docket No. 50-206

References (1) and (2) provided our responses to IE Bulletin 79-06A and Revision 1 thereto which requested information concerning the recent Three Mile Island Incident. Reference (1) addressed IE Bulletin Items 1-12 while Reference (2) addressed IE Bulletin Item 13 and provided schedular information for completing the outstanding action items identified in Reference (1).

The purpose of this letter is to provide additional information to supplement our responses submitted by References (1) and (2). The additional information is contained in the enclosure to this letter entitled, "Supplemental Information to May 3, 1979 and May 23, 1979 Responses to Bulletins 79-06A and 79-06A, Revision 1, San Onofre Nuclear Generating Station, Unit 1, Docket No. 50-206." To facilitate your review of the enclosure the supplemental information has been numbered to correspond to the IE Bulletin action items and our associated responses contained in References (1) and (2).

Should you have any questions or require additional information, do not hesitate to contact me.

Sincerely,

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J T Head

Enclosure
cc: Director, Office of Inspection and Enforcement (12)
Division of Reactor Operations Inspection

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SUPPLEMENTAL INFORMATION TO MAY 3, 1979 AND MAY 23, 1979
RESPONSES TO BULLETINS 79-06A AND 79-06A, REVISION 1
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

DOCKET NO. 50-206

The supplemental information below is numbered to correspond to the Bulletin action items.

Item 2

A thorough review of all transient and accident conditions based on insight gained from TMI-2 is currently in progress. The scope of this review will be sufficiently broad such that subsequent procedure revisions will (a) assure that action steps specifically warn of potential for voiding with a description of all instrumentation which might provide indication of potential or actual voiding, (b) specifically address operator actions, based on operational modes and instrument indication discussed above, for terminating conditions tending to lead to void formation and (c) provide operators with guidance for enhancing core cooling given the unexpected conditions of actual voiding in the primary system.

As stated in our May 23, 1979 letter to the NRC, we have scheduled completion dates of August 15, 1979 for the review and September 15, 1979 for revision of operating instructions based on appraisal of the review. A summary of the results of the review and subsequent procedure revisions will be submitted within two weeks after completion of the review.

Our review thus far has resulted in issuance of station memorandums and procedure revisions which provide operators with instructions and information with respect to identification and control of system

voiding, enhancing core cooling, and maintaining natural circulation capability. We have also identified instrumentation available in the control room which might be utilized in void recognition.

The best means currently available to the operator to determine if voiding has occurred is a comparison of reactor coolant system pressure with reactor coolant hot leg temperature to verify that a margin to saturation conditions exists. As noted in our May 3 response, a curve has been provided to the operators with instructions as to its use and significance.

Other instrumentation available in the control room which might provide the operators with indication of core voiding during natural circulation include:

(1) Incore thermocouples and

(2) Source range nuclear instrumentation

Instrumentation available in the control room to verify natural circulation is being maintained includes:

(1) Reactor coolant hot to cold leg ΔT on all three loops.

(2) Cold leg reactor coolant temperatures on all three loops.

(3) Hot leg and cold leg average reactor coolant temperatures on all three loops.

(4) Incore thermocouples.

In addition, several other indirect indicators of natural circulation would be available in the form of instrumentation and equipment which is indicative of heat removal from the secondary side of the steam generators (i.e., steam generator level and pressure, steam dump, etc.).

Our continuing review of transient and accident conditions may identify further instrumentation which might provide indication of system voiding and/or natural circulation. As noted above, the review results will be summarized and any additional operator aids identified will be reflected in our review description.

In the interim, the above information regarding instrumentation available to detect core voiding and verify natural circulation will be reviewed by station operators by June 30, 1979. Documentation of the review will be maintained in plant records.

Item 4

Our operating procedures require manual initiation of containment isolation based on the existence of conditions described therein which indicate that a loss of coolant has occurred. In addition, manual containment isolation is required if the 2 psig automatic isolation setpoint is reached coincident with safety injection initiation to prevent certain valves from reopening if pressure subsequently drops below 2 psig. Furthermore, we will revise operating procedures as necessary to require manual actuation of all automatic containment isolation valves in the event of automatic initiation of safety injection by July 15, 1979. We are currently evaluating a plant design change which would (1) provide for automatic containment

isolation upon receipt of a SIS actuation signal or 2 psig containment pressure, and, (2) require manual reset of containment isolation. This change would eliminate the necessity for manual operator actions as described above.

The containment isolation signal (CIS) isolates all but the following containment penetrations:

(1) Component cooling water supply and return lines.

(2) Main steam and feedwater lines.

(3) Instrument air supply lines.

(4) RCS Letdown line.

(5) Reactor Coolant Pump Seal Return line.

(6) Lines associated with engineered safety features.

(7) Lines which are normally closed during operating modes requiring containment integrity and remain closed following an accident.

The component cooling water system is completely closed within containment and thus was not provided with isolation valves.

The turbine stop valves serve as isolation valves for the main steam lines. These valves close automatically on turbine trip. The main feedwater lines are isolated via actuation of the Safety Injection system when the feedwater pumps switch to safety injection service.

The instrument air supply line is automatically isolated when supply pressure falls below 60 psig. Since isolation is achieved at a pressure which is above the containment design pressure. (49.4 psig), isolation of this line is provided if the air supply pressure will not assure positive inflow to containment.

The RCS letdown line downstream of the letdown orifices is not designed to accommodate full RCS pressure. Automatic actuation of the containment isolation valves, either during routine CIS testing or inadvertently, would cause lifting of the relief valve which is provided to prevent overpressurization of this line. Therefore, the isolation valves on this line are closed by remote manual actuation when required for containment isolation as specified in our operating procedures.

In addition, the RCS letdown orifice isolation valves are automatically closed upon receipt of a safety injection signal. These valves would isolate the RCS letdown line.

The RCP seal return line is not designed to accommodate full RCS pressure. For the same reasons discussed above for RCS letdown, this line is also provided with remote manual actuation isolation valves which are closed per operating procedure as required.

None of the lines needed for RCP auxiliaries are automatically isolated on CIS or SIS. Therefore, automatic containment isolation does not prevent RCP operation.

Item 7.b

IE Bulletin 79-06A, Revision 1, Item 7.a requested that operators be instructed not to override automatic actions of engineered safety features, unless continued operation of engineered safety features would result in unsafe plant conditions. Item 7.b provided specific instructions for application of the Item 7.a request for automatic safety injection.

Since San Onofre does not have the typical HP/LP safety injection system, our May 3 response to item 7.b required modification to the specific requests to comply with the intent of the criteria therein.

HPI at San Onofre is provided by the charging pumps while LPI at San Onofre is provided by the Safety Injection pumps in combination with the main Feedwater Pumps. Due to the high flow rate of the LPI system, it is possible to pump the design basis quantity of water to the RCS in less than 20 minutes.

Our response to Item 7.b(1) is intended to preclude termination of safety injection (HP and LP) after a LOCA until alignment of the safety injection recirculation system is necessary, regardless of how much time is required to reach this point. For a small break LOCA, this time could greatly exceed 20 minutes while for a large break LOCA, this time could be a minimum of about 8 minutes. In this case, the inventory of water in the Refueling Water Storage Tank has been established to ensure that (1) the design basis quantity of water has been pumped to the RCS

(i.e., within approximately 8 minutes), and, (2) sufficient inventory remains to continue Containment Spray for an additional minimum of 20 minutes. The plant safety analysis was performed based on these assumptions.

Our response to Item 7.b(2) is intended to prevent termination of safety injection (HP and LP) when actuation is not a result of a LOCA for at least 20 minutes and until all hot and cold leg temperatures are and can be maintained at least 50°F subcooled unless the pressure/ temperature criteria for the pressure vessel are reached. These pressure/temperature criteria are contained in our Technical Specification Figure 3.1.3b.

Our May 3 response indicated that Operating Instruction S-3-5.5 provides direction for post-safety injection operator actions. This procedure was revised as noted in the response and included in the operator training program per Bulletin requirements. A thorough review of all emergency operating procedures is being conducted to verify that no specific instructions conflict with the Bulletin requirements. This review will be completed and any identified conflicts will be resolved by June 30, 1979. Any procedures which are modified as a result of this review will be incorporated in the operator training program.

Our review thus far has identified one conflict regarding steam generator tube failure. This accident is a special case LOCA which does not result in utilization of the safety injection recirculation system. In this case, the criteria of 7.b(2) as stated above will be applied until plant cooldown is accomplished and decay heat is being removed through the residual heat removal system.

Item 8

As described in our May 3 response, existing procedures which govern the alignment, alignment requirements and manipulation of safety-related valves during maintenance, testing and plant and system start-up are currently being reviewed. This review is scheduled to be completed by July 31, 1979. The objective of the review is to determine whether or not the procedures adequately ensure that such valves are returned to their correct positions following necessary manipulations. A summary of the results of the review and any procedural revisions necessary will be submitted within two weeks after completion of the review.

As discussed above, the review of procedures is limited to ensuring proper safety-related valve positions following necessary manipulations. During all other operational modes, routine scheduled supervisory periodic surveillance to ensure proper safety-related valve positions is not considered necessary for the following reasons:

1. Position indication for automatic and remote manually operated valves is located in the control room. Should the valve change position, this change of state would be observed in the control room.

2. Manually operated valves important to the proper operation of systems are chained and locked in position, requiring a key and positive intentional action to change the valve position (e.g., locking the valve handwheel or stem prevents the valve from "vibrating" closed or open).

3. There are surveillance requirements incorporated into the Technical Specifications which require equipment or system tests which verify proper valve alignment and/or operability (e.g., monthly hot functional testing of the Safety Injection System, monthly containment isolation valve actuation tests, diesel generator testing, etc).

4. Should a valve not be in the proper position, there are process indications indicative of improper alignment (e.g., loss of flow, pressure increase or decrease, level changes, etc.).

5. Operating personnel perform routine plant inspections which may detect improper valve position (e.g., checking for spills from open valves, discharges from relief valves, etc.).

6. Valves are located in controlled plant areas (both security and radiation) which require preauthorization to enter; thereby limiting access to the valves.

Item 9

Our May 3 response specifically addresses valves returning automatically to original position on resetting of CIS. The first paragraph of our response states that Operating Instruction S-3.5.5 has been revised to require the operator to manually activate the containment isolation signal when he determines that a loss of coolant has occurred. This action is required prior to resetting safety injection and will block the reopening of these valves even if the CIS resets due to containment pressure falling below the 2 psig setpoint.

The design change we are evaluating (see response to Item 4 of this letter) would prevent automatic reset of the CIS, thereby avoiding the requirement for manual operator action.

Item 10

The review and modification of procedures described in our May 3, 1979 letter has been extended to include safety-related systems. The review and modification of procedures is scheduled to be completed by July 31, 1979. A summary of the results of the review and the actions taken will be submitted within two weeks after completion of the review.

Item 10.b

Our May 23 letter listed a scheduled completion date of July 31, 1979 for this item. A summary of the results of our review and the actions taken will be provided within two weeks after completion of the review.

Item 10.c

The Superintendent authorizes removal of systems from service and return to service. Under emergency conditions, this responsibility may be assumed by the Watch Engineer (shift supervisor).

Relay of system status information from one shift to the next is carried out by operating personnel as required by operating procedures.

Each operator finishing his shift discusses all activity accomplished under his cognizance with his counterpart for the oncoming shift (i.e., Watch Engineer with Watch Engineer, Control Operator with Control Operator, etc.). These discussions place emphasis on the status of all

activities in progress at shift change. In addition, the oncoming shift is required to review the Control Operators Log Book for preceeding shifts and to so indicate this review by initialling the book.

Item 11

Station procedures S-E-203 and S-VIII-1.4 have been revised to include an explicit requirement for notification of the NRC within one hour of the time the reactor is not in a controlled or expected condition of operation.

Item 12

Our review and subsequent revision to operating procedures to provide appropriate action for hydrogen control will be expedited to meet a target completion date of July 31, 1979. In the interim, operators have been instructed by a May 3, 1979 memorandum of operating modes for dealing with hydrogen gas.