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January 20, 1983

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Systematic Evaluation Program
Topics III-4.B, IV-2, V-5, VI-2.D, VI-3, VI-4 (Systems), VI-7.A.3, VII-1.A, VIII-3.B
San Onofre Nuclear Generating Station
Unit 1

Your letter dated November 12, 1982 provided information concerning several incomplete safety topics for the San Onofre Unit 1 integrated assessment. In our December 7, 1982 letter, we provided a schedule for submittal of comments or indication that there were no comments on the incomplete topics. Enclosed with this letter are comments (or indication that there are no comments) on eight of the topics for which you requested verification of factual correctness. Comments on the remaining four topics are being forwarded under separate cover. Comments on one additional topic, VIII-3.B, which was not indicated in your November 12, 1982 letter, are also included in the enclosure.

Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection, requires additional study of the sensitivities of the various systems. This study is currently underway and the results will be used during the integrated assessment to assist in determining the need for any modifications.

If you have any questions regarding this information, please contact me.

Yours truly,

R W Krieger

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R. W. Krieger
Supervising Engineer
San Onofre Unit 1 Licensing

Enclosure

Enclosure

SEP TOPIC COMMENTS

Topic III-4.B, Turbine Missiles

Reference: Letter, Walt Paulson, NRC, to R. Dietch, SCE, dated June 24, 1982

Comment

In paragraph 5 on page 3 of the topic evaluation it is stated that the control and stop valve combinations are exercised every two weeks. This frequency is not correct. These valves are exercised monthly while the reactor is in Mode 1.

Topic IV-2, Reactivity Control Systems

Reference: Letter, R. W. Krieger, SCE, to D. M. Crutchfield, NRC, dated November 30, 1982

Comment

Comments on this topic were transmitted in the referenced letter. There are no additional comments at this time.

Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection

Reference: Letter, D. M. Crutchfield, NRC, to R. Dietch, SCE, dated June 3, 1981

Comments

On page 2 of the revised draft evaluation of this topic there are several inaccuracies. Comments on these are as follows:

1. It is stated in No. 1 of the "Evaluation" section that all systems meet the criteria as set forth in Regulatory Guide 1.45. This, however, cannot be concluded from the available information. While different systems for leakage detection do exist as required by the Regulatory Guide, their sensitivities have not accurately been determined. A study is currently underway to determine the time required to achieve the designated sensitivity for these systems.

2. In No. 2 of the "Evaluation" section it is stated that not all systems which interface with the reactor coolant system have been identified. Table 2 is a complete list of systems used during normal power operation that are connected to the RCS. Additional systems connected to the RCS not used for normal power operation are the Safety Injection System, Residual Heat Removal System and Sample System. These systems are normally isolated from the RCS by check valves and automatic valves.

3. In No. 3 of the "Evaluation" section it is indicated that insufficient information concerning the CVCS Makeup Flowrate leakage detection method was provided. This method of RCS leakage detection is an inventory measurement system which measures the time between makeups and the amount of makeup. These inputs are then used to determine total RCS leakage rate. A sensitivity study is currently underway on this system.

4. It should be noted that the data contained in Tables 1 and 2 concerning sensitivity and time required to achieve sensitivity have not been verified. A study is currently in progress to evaluate the sensitivity of these methods. The results will be used during the integrated assessment to determine the need to modify these systems.

5. There is no basis for the information indicated in the column "Earthquake For Which Function Is Assured." The capability of these systems to function following a seismic event will be reviewed when USI A-46, "Seismic Qualification of Equipment in Operating Plants" is finalized.

6. Too much has been inferred from SCE's letter dated April 8, 1981 which should not be used in the Table 1 column "Documentation Reference." As indicated in comment No. 4 above, information concerning sensitivity and time required to achieve such will be presented during the integrated assessment.

7. Table 1 indicates that all leakage detection systems at San Onofre Unit 1 are testable during normal operation. This is not true. The airborne particulate and gaseous radioactivity monitors are testable and the sump pump actuations are monitored by administrative procedure (not time meters as stated). The other systems are not testable during normal operation.

8. Due to the comments presented above, the conclusion that the leakage detection system incorporated for measurement of leakage from the reactor coolant pressure boundary to the containment are in conformance with Regulatory Guide 1.45 cannot be made. The sensitivities of the systems are currently being evaluated and the results of this study will be used during the integrated assessment to determine the need for modifications to these systems.

Topic VI-2.D, Mass and Energy Release for Possible Pipe Break Inside Containment

References: A. Letter, D. M. Crutchfield, NRC, to R. Dietch, SCE, dated January 12, 1982

B. Letter, K. P. Baskin, SCE, to D. M. Crutchfield, NRC, dated November 18, 1982

C. Letter, K. P. Baskin, SCE, to D. M. Crutchfield, NRC, dated November 4, 1981

D. Letter, K. P. Baskin, SCE, to D. M. Crutchfield, NRC, dated June 30, 1982

Comments

1. On page 11 of the safety evaluation enclosed with the Reference A letter it is indicated that peak containment pressure and temperature are 53.0 psig and 404^{o}F, respectively. As indicated in the Reference B letter, the revised Temperature and Pressure for the Main Steam Line Break have been recalculated using 8% revaporization of condensate in a superheated atmosphere per NUREG-0588, Appendix B. The revised temperature and pressure are 391.5^{o}F and 53.3 psig, respectively.

2. In the Reference C letter it was indicated that the double ended guillotine main steam line break is not a credible event. In Reference D the results of a more probable main steam line rupture analysis was provided. This analysis demonstrated that the LOCA anaysis was a bounding event and will be used in the qualification of existing equipment as part of the SCE Environmental Qualification program.

Topic VI-3, Containment Pressure and Heat Removal Capability

Reference: Letter, D. M. Crutchfield, NRC, to R. Dietch, SCE, dated January 12, 1982

Comment

SCE has no comments on this topic.

Topic VI-4 (Systems), Containment Isolation System

References: A. Letter, Walter A. Paulson, NRC to R. Dietch, SCE, dated December 6, 1982

B. Letter K. P. Baskin, SCE to D. M. Crutchfield, NRC, dated November 22, 1982

Comment

The Reference B letter submitted SCE's comments on the draft safety evaluation of the subject topic. Reference A enclosed a revised safety evaluation reflecting the comments of Reference B. There are no additional comments at this time.

Topic VI-7.A.3, ECCS Actuation System

Reference: Letter, D. M. Crutchfield, NRC, to R. Dietch, SCE, dated November 18, 1981

Comments

1. In Section 3.1 of the EG&G Idaho report enclosed with the reference it is stated that "Two-out-of-three low pressurizer pressure and two-out-of-three high containment pressure signals will initiate the operation of both subsystems." The statement should state that two-out-of-three low pressurizer pressure or two-out-of-three high containment pressure signals will initiate the operation of both subsystems. It should be further clarified that either of the two logic trains will actuate the safety injection system. Manual initiation also is an option.

2. In Section 3.2 of the EG&G Idaho report a description of the tests and surveillance for the safety injection system is put forth. Since the time this report was prepared, revision of the safety injection system surveillance program has occurred. The applicable sections of the San Onofre Unit 1 Technical Specifications are Sections 4.2.1, "Safety Injection and Containment Spray System Periodic Testing," and 4.2.3, "Safety Injection System Hydraulic Valve Testing (Surveillance Requirement)." In summary, these sections require, among other things, the following:

    a. At intervals not longer than normal refueling, a "no-flow" safety injection system test,

b. At intervals not longer than normal refueling, a "no-flow" containment spray system test,

c. Component tests of the safety injection pumps and recirculation pumps at intervals not exceeding one month during periods when the reactor is critical,

d. Component tests of the refueling water and spray additive pumps at intervals not exceeding one month when the reactor coolant system temperature is above 200°F,

e. At 92 day intervals while in Modes 1 through 4 (see Technical Specification 4.2.3 for more details on the intervals required) a hot SIS functional test is performed.

In addition to the above tests, the refueling water storage tank inventory and boron concentration are specified in Technical Specification 3.3.3 as correctly noted in the EG&G Idaho report. Boron concentration is checked monthly while Refueling Water Tank Hi-Lo Level is annunciated in the control room.

3. Section 4.1 on page 4 states that the recirculation system "injects coolant from the containment sphere sump into the reactor coolant system via a line from the reactor coolant pump seal water injection lines." Recirculation occurs through three lines from the reactor coolant pump seal water injecton lines to the cold leg injection lines.

4. In Section 4.1 it is stated that "the refueling water pumps also draw water from the heat exchanger and can serve as backup components." This statement may be misleading as it is not according to normal procedure to use these pumps as backup. Their primary function during and after a LOCA is to drive the sphere spray system. However, connections do exist between the normal containment spray system and the recirculation injection system so that they can be used as a backup in an abnormal situation.

5. Also in Section 4.1 it is stated that "The refueling water pumps are started automatically on a safety injection signal." This statement is not correct. The refueling water pumps are used as an integral part of the sphere spray system during and following a LOCA. They are actuated by a "containment spray actuation signal" generated through an "and" configuration by high containment pressure and safety injection actuation signals.

6. It is further stated in Section 4.1 that "Suction for the [recirculation or refueling water--it is not clear] pumps can also be from the refueling water storage tanks." Again, though factually correct (except that there is only one refueling water storage tank and not tanks as stated), suction would not normally be from the refueling water storage tank as this source would presumably be depleted during the primary injection phase during a safety injection. It could however, be replenished and suction reestablished.

7. On page 5 of the EG&G report in the same Section (4.1) it is stated that "The recirculation system, except for the refueling water pumps, has no automatic sequencing." The refueling water pumps are only automatically sequenced for their function as containment spray pumps. As they are not normally included as part of the safety injection recirculation system and the portion of the statement referring to these pumps is not relevent. The corrected statement should therefore be: "The recirculation system has no automatic sequencing." In the statement following it is stated that the system is actuated by the operator when safety injection flow becomes low and the containment sphere sump level is high enough to support the recirculation system. This is not completely true. The revised statement should be as follows: The system is actuated by operator action when the refueling water storage tank level becomes low or the containment sphere sump level is high enough to support the recirculation system.

8. On page 5, in the second paragraph on Section 4.2 it is stated: "Therefore, it would be possible to leak radiation into the component cooling water system when the recirculation system is needed. As this heat exchanger is the only heat exchanger for the recirculation system as well as the containment spray system, it could not reasonably be valved out if the leak were detected when its operation was required." This portion of the paragraph should be clarified as follows: "Therefore, it would be possible to leak radioactive material into the component cooling water system without detection. This heat exchanger is the only heat exchanger for the recirculation system as well as the containment spray system. When its operation is required, it could not reasonably be valved out even if the secondary water was found to be radioactive.

Summary

The summary of the EG&G Idaho report and the conclusions in the staff's safety evaluation report should be modified to reflect the above information. Specifically, with the additional information supplied it can be concluded that the safety injection system is periodically pressure and functional tested under conditions as close to design as practical. All ECCS pumps are periodically tested and the tests performed confirm the operability of the actuation circuitry.

Topic VII-1.A, Isolation of Reactor Protection System from Non-Safety Systems, Including Qualification of Isolation Devices

Reference: September 9, 1982 letter from D. M. Crutchfield, NRC, to R. Dietch, SCE

Comments

1. The table of RPS parameters and associated trip logic in Section 3.1 of your contractor's topical evaluation report should be revised in the following manner:

a. High Flux Level Trip Logic is 2 out of 4 rather than the stated 1 out of 4.

b. Safety Injection Reactor Trip Logic is 1 out of 2 rather than the stated 1 out of 4.

2. Section 3.1.3 of your contractor's topical assessment states:

"The steam and feedwater flow sensor also transmits a signal directly to the Optimac computer which operates the level control system for the three steam generators by controlling the feedwater flow. The same input signals for the steam generator control are also transmitted to process recorders YR456, YR457 and YR458. Contacts on these recorders initiate annunciator actions."

Contacts on these recorders no longer initiate annunciator actions. Annunciator actions are now initiated by an action pak that receives voltage input from the steam and feedwater flow sensor. The action pak is hooked up in a parallel configuration with the process recorders and allows precise selection of the actuation voltage.

3. Section 3.1.4 of your contractor's topical assessment states:

"Auxiliary contacts from the scram relays initiate annunciator action while instrumentation separate from the above RPS system provides status monitoring."

The reactor coolant flow protection system is of similar design to that of the pressurizer pressure and pressurizer level protection systems. The pressurizer level and pressurizer pressure indicators are not isolated from their respective trip bi-stables as stated in Sections 3.1.1 and 3.1.2. Similarly, the reactor coolant flow indicator is not isolated from its associated trip bi-stable. This consideration may influence the Evaluation in Section 3.1.4.

4. Section 3.1.7 of your contractor's topical assessment report states:

> "Other outputs from the intermediate range nuclear instrumentation include signals to local and remote indicators, a recorder and the data logger."

The referenced data logger does not receive output signals from the intermediate range nuclear instrumentation. This data logger is disconnected and is not in use at San Onofre Unit 1. Therefore, the evaluations contained in Section V, Item 2 of your safety evaluation report and Sections 3.1.7, 3.1.8 and 4.0, Item 2 of your contractor's topical assessment report should incorporate this consideration.

Topic VIII-3.B, dc Power System Bus Voltage Monitoring and Annunciation

References: A. January 28, 1981 letter from D. M. Crutchfield, NRC, to R. Dietch, SCE

B. July 24, 1981 letter from D. M. Crutchfield, NRC, to R. Dietch, SCE

Comments

Section 3.2 of your contractor's topical assessment states in part:

> "The San Onofre 1 control room has no indication of battery current, charger output current, bus voltage, charger output voltage, battery high discharge rate, bus undervoltage (UPS), bus overvoltage, bus ground (UPS), battery breaker status (UPS) or charger output status."

Section V of your safety evaluation states in part:

> "As noted in EG&G Report 1357F the San Onofre control room has no indication of battery current, battery charger current, dc bus voltage, or breaker/fuse status."

These statements are partially in error as the control room does have the following indicators and alarms:

a. Bus #1 voltage indicator;

b. battery charger failure alarm for chargers associated with dc Busses 1 and 2; and

c. UPS failure alarm.

The safety evaluation report should incorporate these considerations.

GEH:6771