April 29, 2013

Mr. Peter T. Dietrich
Senior Vice President and Chief Nuclear Officer
Southern California Edison Company
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, CA 92674-0128

SUBJECT: SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 – RELIEF REQUEST ISI-3-36 PRESSURE TEST, REQUEST FOR RELIEF FROM INSERVICE INSPECTION REQUIREMENTS FOR THE DURATION OF THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NOS. ME8486 AND ME8487)

Dear Mr. Dietrich:

By letter dated April 17, 2012 (Agencywide Documents and Access System (ADAMS) Accession Number ML12109A079), as supplemented by letter dated November 15, 2012 (ADAMS Accession Number ML123210179), Southern California Edison Company requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for pressure testing.

Specifically, the licensee requests authorization of an alternative to the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-P, Item Number B15.50 and B15.70 for the system leakage test of Class 1 Reactor Coolant System piping.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review and concludes in the enclosed safety evaluation that the proposed alternative provides reasonable assurance of structural integrity and leak tightness, and that complying with the specified ASME Code, Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative for San Onofre Nuclear Generating Station, Units 2 and 3 for the duration of the third 10-year Inservice Inspection interval.

If you have any questions, please contact Brian Benney, at (301) 415-2767, or via e-mail, at Brian.Benney@nrc.gov.

Sincerely,

Douglas A. Broaddus, Chief
SONGS Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION PROGRAM REQUEST FOR RELIEF NO. ISI-3-36

SAN ONOFRE NUCLEAR GENERATING STATIONS, UNITS 2 AND 3

SOUTHERN CALIFORNIA EDISON

DOCKET NOS. 50-361, 50-362

1.0 INTRODUCTION

By letter dated April 17, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12109A079) as supplemented by letter dated November 15, 2012 (ADAMS Accession No. ML123210179), San Onofre Nuclear Generating Station (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for pressure testing.

Specifically, the licensee requests authorization of an alternative to the requirements of ASME Section XI, Table IWB-2500-1, Examination Category B-P, Item Number B15.50 and B15.70 pursuant to *Title 10 of the Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(ii), which states that compliance with the specified requirement would result in hardship without a compensating increase in the level of quality and safety. The submitted Relief Request ISI-3-36, for Units 2 and 3 for the third 10-Year inservice inspection (ISI) interval, pertains to performance of a system leakage test of Class 1 Reactor Coolant System piping.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for ISI of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that ISI of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12-months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

Section 50.55a(a)(3) of 10 CFR states, in part, that alternatives to the requirements of paragraph 50.55a(g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if (i) an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if (ii) the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above analysis, the staff finds that regulatory authority exists to authorize an alternative to the ASME Code requirement, as requested by the licensee.

Enclosure

3.0 TECHNICAL EVALUATION

3.1 Relief Request ISI-3-36

3.1.1 System/Components(s) for Which Relief is Requested

Class 1 pressure-retaining components, examination category B-P, Item Numbers B15.50 and B15.70.

3.1.2 Applicable Code Requirement

The 1995 Edition, 1996 Addenda of ASME Section XI, Table IWB-2500-1, Examination Category B-P, Item Numbers B15.50 and B15.70 requires all Class 1 pressure retaining components be subject to a system leakage test with a VT-2 visual examination in accordance with IWB-5220. This pressure test is to be conducted prior to plant startup following each reactor refueling outage. The pressure retaining boundary for the test conducted at or near the end of each inspection interval shall be extended to all Class 1 pressure retaining components within the system boundary per IWB-5222(b).

3.1.3 Basis for Relief (as stated)

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The many vent, drain, and branch (VTDB) connections 1-inch nominal pipe size (NPS) and smaller off the reactor coolant pressure boundary and four 2-inch drain lines from cold leg to reactor coolant drain tank have double manual isolation valves.

The requirement to extend the system leakage test boundary for the leakage test conducted at or near the end of each inspection interval to the outboard valve on these VTDB connections results in a hardship without a compensating increase in the level of quality and safety. Repositioning the inboard manual valves before and after the test will take considerable time and will result in an unnecessary increase in dose to plant personnel. Manual operation of the VTDB valves is estimated to expose plant personnel to 1.2 man-rem per test of all required valves. These off-normal configurations may also contribute to the risk of delaying normal plant startup because of the critical path time and effort required to ensure system configuration is restored.

The purpose of the required extended pressure boundary condition is to detect evidence of leakage resulting in a validation of the integrity of the Reactor Coolant System (RCS) pressure boundary beyond the first isolation valve. While in Mode 3 during plant startup, the system leakage test is performed with the RCS at full pressure, approximately 2250 pounds per square inch absolute (psia) at elevated temperature. Testing these VTDB connections would require an operator to change valve positions with the RCS at 2250 psia. Furthermore, due to the location of these valves, it would be necessary to erect scaffolding for this evolution. Finally, in Mode 3 during plant startup the system leakage test is conducted as a critical path evolution. The valve manipulations necessary to pressurize the isolated portions of the vent, drain and instrument connections, and then to return them to normal position would directly impact the startup activity sequence and outage duration. Meeting those requirements involves considerable time to establish and return from the required temporary configuration resulting in both risk of delaying normal plant startup following a refueling outage and an increase in personnel radiation exposure and industrial safety risk.

In the November 15, 2012, letter, the licensee goes on to say that there is an inherent industrial safety risk when performing work in congested areas, in close proximity to valves and piping at increased pressures and temperatures. The VT-2 test is performed during startup with RCS pressure at or above 2250 psia and 280 degrees Fahrenheit. Personnel in the area during the test include operations personnel, VT-2 examiners and personnel erecting and dismantling the scaffolding.

3.1.4 Proposed Alternative (as stated)

Reactor coolant pressure boundary VTDB connections 1-inch NPS and smaller and four 2-inch drain lines from cold leg to reactor coolant drain tank will be visually examined for leakage with the inboard isolation valve in the normally closed position during the system leakage test conducted at the end or near the end of each inspection interval per IWB-5222(b). This test provides reasonable assurance of structural integrity.

The 1-inch NPS and smaller and four 2-inch drain lines from cold leg to reactor coolant drain tank VTDB connections are normally closed during plant operation. The outboard valves would only see pressure if the inboard valve is open or leaks by the seat. Seat leakage, although undesirable, is not indicative of a flaw in the pressure boundary. The non-isolable portion of these VTDB connections is pressurized and VT-2 visually examined during the test. The VT-2 visual examination performed each refueling outage extends to the outboard valve, even though it is not pressurized. SONGS Units 2 and 3 technical specifications for reactor coolant pressure boundary leakage monitoring requires appropriate actions, including plant shutdown if leakage exceeded specified limits.

3.2 NRC Staff Evaluation

The subject segments are lines and connections equipped with manual valves providing double isolation of the RCS. Repositioning the inboard manual valves before and after the test will take considerable time and will result in an increase in dose to plant personnel. The licensee has stated that the non-isolable portion of the RCS boundary lines and connections will be pressurized and VT-2 examined as required. The isolable portion will have a VT-2 examination while the line is pressurized to the extent possible. These lines and connections do not have test connections that would allow them to be individually pressure-tested without design modifications. Pressurization would require an inboard valve to be opened, which removes the double isolation feature and is a safety hazard for the personnel performing the tests, due to high temperature and pressure.

In addition, the VTDB lines and connections are not subject to high or cyclic loads. The NRC staff finds that the licensee's proposed alternative would provide reasonable assurance of structural integrity while achieving as low as reasonably achievable goals and that compliance to ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the proposed alternative provides reasonable assurance of structural integrity and leak tightness, and that complying with the specified ASME Code, Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative for San Onofre Nuclear Generating Station, Units 2 and 3 for the duration of the third 10-year ISI interval.

All other requirements of the ASME Code for which relief has not been specifically requested and authorized remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Margaret Audrain

Date: April 29, 2013

If you have any questions, please contact Brian Benney, at (301) 415-2767, or via e-mail, at Brian.Benney@nrc.gov.

Sincerely,

/RA/

Douglas A. Broaddus, Chief
SONGS Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-361 and 50-362

Enclosure:
Safety Evaluation

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