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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Gentlemen:

SUBJECT: **Docket No. 50-206**
Reactor Vessel Support Embrittlement,
Plant Specific Applicability Evaluation of NUREG/CR-5320
San Onofre Nuclear Generation Station, Unit 1

The results of our evaluation of the potential for reactor vessel support embrittlement indicate that San Onofre Nuclear Generating Station Unit 1 (SONGS 1) is not susceptible to this condition. The subject of low radiation, low temperature embrittlement of the reactor vessel supports is discussed in NRC Generic Issue 15 and NUREG/CR-5320. Based on our evaluation, the calculated stress levels at SONGS 1 are a factor of two lower than those calculated in NUREG/CR-5320, and do not indicate that support embrittlement is of concern at SONGS 1 for the licensed life of the Unit.

BACKGROUND

Reactor vessel ebmrittlement was identified as an open item in the NRC Order dated January 2, 1990 (TAC No.M11232). The issue was subsequently evaluated in NUREG/CR-5320 for two plants, Trojan and Turkey Point. We reviewed NUREG/CR-5320 and performed a plant specific evaluation for SONGS 1 to resolve this issue.

Radiation embrittlement is of concern because it increases the potential for propagation of any pre-existing flaws in the reactor support structures. This could potentially lead to eventual support failure, provided the following conditions exist in the support structure:

1. A critical portion of the reactor support is located in an area of relatively high neutron flux, such as, a location within the core length section.

2. The support critical section is under a sustained tensile stress.

3. Flaws of critical size already exist in the support material.

The NUREG/CR-5320 study utilized data from the High Flux Isotope Reactor (HFIR) vessel surveillance program at Oak Ridge National Laboratory (ORNL),

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which suggested a potential for radiation embrittlement at low temperature and low fluence. The result of the NUREG analyses indicated that a critical crack depth as small as 0.2 inch to 0.4 inch could be of concern after 32 Effective Full Power Years (EFPY) of operation. The analysis also concluded that low cycle fatigue is not a viable mechanism for creation of flaws of this size. Therefore, such flaws would have to exist at the time of fabrication.

Prior to the issuance of the NUREG, the Westinghouse Owners Group (WOG) was requested by the NRC to review the NUREG. The WOG concluded that the NUREG analysis was considerably conservative due to several reasons, including:

1. Extrapolation of the HFIR irradiation results to the reactor supports of PWR plants.

2. Use of upper bound faulted loads instead of best estimate loads, under normal operating conditions.

The WOG provided specific plant information on the reactor vessel support designs, fabrication and operating environments for 50 plants including SONGS 1. The WOG report concluded that, based on the collected information, the worst case support design and fabrication condition has already been analyzed in the NUREG. The results of the WOG analysis showed that no short term concerns exist regarding embrittlement of the reactor support. To evaluate the susceptibility of the SONGS 1 reactor vessel support to this embrittlement condition, we conservatively elected to perform a calculation.

METHODOLOGY

The following steps were utilized in our evaluation:

1. The potential stresses in the SONGS 1 reactor vessel support were calculated based on its specific geometry, and assuming faulted loads. (NOTE: The SONGS 1 vessel support is a vertical type, whereas the supports in NUREG/CR-5320 are the cantilever (horizontal) type. The stresses which result from the vertical support geometry are lower than those associated with a cantilever/horizontal type support because the cantilever type has a larger bending moment.)

2. The stress intensity factor was calculated using the stresses calculated in step 1 and assuming a crack size of 0.5 inch. (NOTE: A crack size of 0.5 inch is conservatively higher than the values in NUREG/CR-5320.)

3. The calculated stresses and stress intensity factor for SONGS 1 were compared to NUREG/CR-5320. The values for SONGS 1 were more conservative than NUREG/CR-5320 by a factor of two.

RESULTS/CONCLUSIONS

As a result of our evaluation, we concluded that the reactor vessel support at SONGS 1 is not susceptible to radiation embrittlement damage for the following reasons:

1. The SONGS Unit 1 reactor vessel supports are of the vertical type. Most of the support is under a sustained compressive stress. Only the top section, which is under bending, has some tensile stress.

2. The top section of the vessel support is located in an area above the core where the radiation level is lower than it is in the core length area.

3. The calculated maximum tensile stress for SONGS 1 is relatively low (9 Ksi). In the unlikely event that a 0.5 inch deep fabrication crack exists, the calculated stress intensity factor (14.75 Ksi$\sqrt{in}$) is significantly below the lower limit (31 Ksi$\sqrt{in}$) calculated in NUREG/CR-5320. Therefore the SONGS 1 design is more conservative by a factor of two (31/14.75=2.1).

4. No flaws exceeding the ASME Section XI Code allowables were detected in the integral welds of the brackets during the recent Cycle 11 outage inservice inspection.

5. There is no recorded evidence of any flame cutting in the reactor support and all bolted connections are friction type, shop drilled, and bolted.

If you have any question regarding this matter, or desire additional information, please contact me.

Very truly yours,

R M Rosenblum

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