UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 TO LICENSE NO. DPR-13

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE UNIT NO. 1

DOCKET NO. 50-206

Introduction

During the refueling and maintenance outage which began on September 15, 1978, Southern California Edison Company (the licensee) had completed refueling of the San Onofre, Unit 1 (SO-1) reactor for Cycle 7 operation and other tasks. The licensee has successfully completed an acceptable steam generator inspection program. This Safety Evaluation (SE) documents our review of these matters in support of the proposed license amendment, which authorizes Cycle 7 operation of SO-1, with appropriate changes to the facility Technical Specifications. This SE also evaluates other Technical Specifications proposed by the licensee, to satisfy NRC staff positions, stated in our SE issued on April 1, 1977 with Amendment No. 25, in connection with the completed Sphere Enclosure Project. With the agreement of the licensee we have made some modifications to the proposed technical specifications.

A. Cycle 7 Reload

Discussion

By application dated August 18, 1978 (Reference 1), as supplemented by letter dated September 22, 1978 (Reference 2), the licensee proposed changes to the SO-1 technical specifications to permit Cycle 7 operation.

For the proposed reload of Cycle 7, one Region 2 and 51 Region 6 fuel assemblies will be removed and replaced by 52 region 9 fuel assemblies. As in previous cycles, stainless steel fuel rod cladding will be used. The licensee has proposed to change its fuel management. A strict three fuel batch "out-in" fuel management scheme will be replaced by a three fuel batch hybrid "in-out" pattern.

The analyses supporting Cycle 7 operation are based on assumed nominal operation at 1347 Mwt, 2100 psia system pressure, 553 F core inlet temperature, and 4.64 kw/ft average linear heat generation rate.

The licensee's proposed changes to the Technical Specifications would:

(1) Replace the description of the fuel management pattern with a more general description, and

(2) Change the bases section of the rod insertion limit technical specification to reflect reanalysis of the ejectec rod hypothesized accident using higher values of post ejected, peak linear heat generation rate.

Evaluation

Fuel Design

The licensee has stated (Reference 1) that the fuel design for Region 9, the fresh fuel, is the same as Region 8 fuel which was used in the previous cycle, with the exception of a modification of the hold-down spring package. This package has been modified to employ four, rather than two, leaf springs. In response to our request for additional information, (Reference 3), the licensee provided (Reference 2) a summary evaluation of the design modification under seismic and blowdown forces. The licensee concluded that the increased hold-down force of the modified design will decrease the fuel assembly impact force on the top core plate during seismic and blowdown conditions. The analysis was performed for vertical seismic forces consistent with the SO-1 DBE (Design Basis Earthquake). This modification has been approved and utilized in other Westinghouse cores. Abnormalties due to the spring modifications have not been observed. On this basis, we accept the licensee's conclusion.

The fuel design (selection of initial backfill pressure) was based on the Westinghouse revised design criteria (Reference 4) and evaluation model (Reference 5). The design criterion has been revised from a restriction that the fuel rod internal pressure not exceed system pressure during Condition I and II events to less restrictive criteria that, "The internal pressure of the lead rod in the reactor is limited to a value below that which could cause (1) the diametrical gap to increase due to outward cladding creep during steady state operation, and (2) extensive DNB propagation to occur." The revised criteria have been approved by us, the evaluation model has not.

The revised Westinghouse fuel performance analysis code, (Reference 5) PAD 3.3, which models enhanced fission gas release at high burnups, was used to show conformance to the deisgn basis. This code, is considered by Westinghouse to provide a conservative estimate of fuel performance. PAD 3.3 is currently understaff review, which is near completion, and is expected to be found acceptable with modification.

The licensee states that the minimum burnup at which the fuel diametrical gap is predicted to increase is in excess of 55,000 MWD/MTU. The licensee has predicted the peak rod burnup for the SO-1 Cycle 7 core to be < 45,000 MWD/MTU. Thus, there is margin to accommodate modifications to PAD 3.3, which will be required by the staff. Based on our current review, we believe that this margin is sufficient to accommodate the required modification.

Clad flattening is not predicted to occur during Cycle 7. The licensee has predicted that all fuel regions of the core have a clad flattening time equal to or greater than 50,000 EFPH. No fuel region is predicted to receive this exposure.

Reactor Design

Core Loading

The licensee has proposed to alter the fuel loading pattern. The current Technical Specifications which describe the core in detail is to be replaced by a general description. This proposal is acceptable.

Previous cores have consisted of three fuel batches with fresh fuel located on the core periphery and once and twice burnt fuel batches loaded in a checkerboard pattern in the interior of the core. The proposed pattern (Reference 1, Figure 2) loads most fresh fuel on or near the core periphery with selected fresh fuel assemblies located in the interior of the core, and most once and twice burnt fuel assemblies loaded in the interior of the core with selected burnt fuel assemblies on the core periphery. A strict checkerboard pattern, by fuel batch in the interior of the core has not been preserved.

The new pattern is typical of those currently employed at several reactors. Judicious selection of individual fuel assembly location in the new pattern reduces the core neutron leakage and results in fuel savings.

Control Strategy

The current ECCS analysis is applicable for values of the peak to average linear heat rate, Fq, times the fraction of rated power, P, less than or equal to 2.95 (Reference 6). Past operation of SO-1 was permitted at values of Fq x P $\leq$ 3.23 based on previous LOCA analyses. This change compensated for the effects of potentially reduced safety injection flow in the event of a large and small size break LOCA, or steam line break (Reference 6, 7) and has been previously reviewed by the staff. The permitted value of Fq is, even at the reduced value, considerably greater than permitted at other Westinghouse plants (typically 2.32). Hence neither constant axial offset control, CAOC, nor the Westinghouse Axial Power Distribution Monitoring System need be employed. The plant is controlled to an envelope of axial offset versus power level and permitted control rod insertion versus power level. Within these envelopes, the licensee has shown (Reference 1, Figure 2) that during condition I operation (steady state and load follow operation) the peak linear heat rate limiting condition for operation will not be violated. The supporting analyses were performed by Westinghouse for the licensee using the methods described in WCAP 8130/8131 (Reference 8) which have previously been accepted. It is noted that expected values of Fq x P are near 1.8, i.e., significantly less than the limiting condition for operation.

Shutdown Margin

The fuel vendor's predicted shutdown margin (scrambled control rod reactivity worth less shutdown requirements) from hot full power beginning and end of Cycle 7 is 2.2% $\Delta\rho$. The required shutdown margin to accommodate the current Steam Line Break accident analyses at the beginning and end of Cycle 7 are 1.25 and 1.9% $\Delta\rho$ respectively. Hence, there is a small predicted excess margin at end of Cycle 7, viz. 0.3% $\Delta\rho$. No excess margin was demonstrated in Cycle 6. Alteration of the fuel management resulted in a 0.1% $\Delta\rho$ increase in the predicted scramble reactivity (all control rods inserted, less worst stuck rod, less 10% calculational uncertainty). Another 0.2% $\Delta\rho$ margin was shown by reduction in the conservatism of the assumed reactivity defect.

Control rod reactivity worth and the isothermal temperature coefficient at hot zero power beginning of cycle will be measured as part of the cycle startup tests. Meeting review criteria associated with these tests will further support the licensee's assertion that the reactor will exhibit adequate shutdown margin at end of Cycle 7.

Kinetics Characteristics

The fuel vendor has predicted values of reactivity coefficients, with the exception of the most negative doppler coefficient, to be within the bounds of values used in the safety analyses. The values used in the safety analysis are extreme rather than expectation values. There is no basis to suspect that the reload fuel will sufficiently perturb the reactivity coefficients such that they are outside the bounds of the extreme values used in the safety analysis.

The most negative doppler coefficient used in the safety analysis is more negative than used in previous analyses. The change is in a conservative direction. As above, this is a change of an extreme value used in the safety analysis. The actual or expection value is a strong function of fuel temperature and a weak function of core burnup, and the details of the fuel management (and hence spatial flux distribution and importance weighting). The use of pre-pressurized fuel results in lower predicted fuel temperatures and in turn a more negative predicted doppler coeeficient.

A slower trip reactivity insertion rate was used by Westinghouse in the Cycle 4 safety analysis. The change is the result of revision of the fuel vendors modeling rather than the result of the reload. Specifically a more bottom skewed axial flux distribution was used to calculate the trip reactivity as a function of core height. Assumed rod positions versus time after scram initiation have not been altered. We have concluded that this change is in a conservative direction and is therefore acceptable.

Accident Analysis

The Cycle 7 accident analysis methods are enumerated in the Westinghouse Reload Safety Evaluation Methodology topical report, WCAP-9272 (Reference 9). Exceptions are discussed below. This report is to a large extent a compendium of methods and computer codes which have been individually reviewed by the NRC staff. Some new material, particularly concerning data links was introduced. The staff is currently reviewing this report. Review has progressed sufficiently to warrant use of the report as a reference document.

All hypothetical accidents originally analyzed in the FSA (Reference 10) were reviewed by the fuel vendor. The licensee stated (Reference 1) that "In most cases, it was found that the effects (of the reload) were accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis." Based on the review of the Cycle 7 kinetics parameters and shutdown margin, and the fact that the fuel assembly design (with the exception of the holddown springs) and the limiting safety system setpoints have not been altered, the staff accepts this conclusion.

Loss of Coolant Accidents

The Loss of Coolant Accident was recently reanalyzed (Reference 6) and approved by the staff (Reference 7), see Control Strategy.

Steamline Break Accident

The Steamline Break accident was not reanalyzed for Cycle 7. However, the methods used in the reference analyses which predicted a minimum DNBR greater than 1.30 are currently being generically reviewed by the NRC. The hypothetical steamline break is a design bases event for which limited clad failure is permitted. If clad failure was to occur the release path available would be through the steam generators which would be assumed to be leaking at the maximum rate permitted by the Technical Specification. Because of this tortuous leak path, a significant number of the fuel rods could be failed without exceeding the site boundary dose limits. The relative power density predicted during the course of a steamline break with all control rods, except the most reactive rod inserted, is highly non-uniform. The predicted minimum DNBR during the transient would occur near the region of the stuck control rod and would be restricted to a small region of the core. Even if departure from nucleate boiling did occur, and even if clad failure did occur, it is the NRC staff judgement that a relatively small fraction of the fuel rods would fail and hence site boundary dose limits would not be violated. On this basis, operation during Cycle 7 is acceptable.

Loss of Flow Accident:

The Loss of Flow Accident assuming the revised scram reactivity insertion rate was reanalyzed. The fuel vendor concluded that the minimum DNBR met the design basis of 1.3. The accident was analyzed assuming the flow coastdown shown in the SO-1 FSA instead of a coastdown calculated using the PHOENIX code. Based on the relative insensitivity of the gross core behavior during this hypothetical accident to cycle specific input values and prior review of the Loss of Flow accident, we accept this conclusion.

Ejected Rod Accident:

The beginning of cycle, hot full power and hot zero power, ejected control rod accidents were reanalyzed by the vendor and reflect increases in predicted values of post ejected peaking factors. The control rod reactivity worths assumed in the analyses were not altered. The changes are of the magnitude associated with a typical reload. Previously approved analytical methods were used (Reference 11). Predicted results of the Westinghouse analysis are acceptable.

Uncontrolled Rod Withdrawal at Power and Uncontrolled RCCS bank Withdrawn from Subcritical Accidents:

These accidents were reanalyzed by the fuel vendor using the revised slower trip reactivity rate, and for the control rod withdrawal at power the revised doppler coefficient was used. Only the maximum reactivity insertion associated with the RCCA (Rod Cluster Control Assembly) Withdrawal at Power was examined to determine the sensitivity of this accident to the change in the trip reactivity function. The RCCA withdrawal from Subcritical was analyzed assuming reactor trip on the power range high neutron flux high setting rather than the low setting stated in WCAP-927?. This is a conservative assumption. Westinghouse found the effect of these revisions to be small and predicted the minimum DNBR to remain above 1.30. These assertions are consistent with our expectations and on this basis the analysis is considered acceptable.

Loss of Load

The loss of load anticipated operating occurence was reviewed by Westinghouse for the licensee. The licensee stated (Reference 1) that the additional energy generated during the loss of load occurence due to the revised trip reactivity function could be accommodated by exisitng margins in the analysis. Based on the relative insensitivity of the loss of load occurrence to the trip reactivity function, we accept this conclusion.

Technical Specifications

Technical Specification 3.5.2, Control Group Insertion Limits

The licensee has proposed to modify the bases section to reflect reanalysis of the ejected control rod accident at beginning of cycle, hot full-power and hot zero-power, with higher values of the post ejected peak linear heat rate. The change is acceptable (see Accident Analysis).

Technical Specification 5.3, Reactor

The licensee has proposed to delete the description of the "out-in" refueling pattern. A more general description of the core is to be substituted (see Reactor Design, Core Loading). The change is acceptable.

Startup Testing

The licensee has described the physics startup test program for Cycle 7 (Reference 12). This program includes the test acceptance criteria and the actions to be taken if the acceptance criteria are not met.

The program includes critical boron concentration measurements, rod group reactivity worth measurements, isothermal temperature coefficient measurements, boron reactivity worth measurements and low power core map. Power ascension tests will consist of core maps at full power. The licensee will submit a summary of the test results to the NRC within 45 days of completion of the test program. The physics startup test program has been reviewed by us and found to be acceptable.

B. Steam Generator Inspections and Proposed Technical Specifications

Discussion

Our Safety Evaluation (SE) issued on April 20, 1978, in conjunction with Amendment No. 20 includes a comprehensive summary of steam generator operating experience at SO-1. In that SE we stated that the performance of subsequent steam generator inspections on a refueling outage frequency may be justified if results of steam generator inspections performed during the Cycle 7 refueling outage indicate that tube denting and support plate cracking has been arrested. By letter dated August 17, 1978, the licensee submitted its "Steam Generator Inspection Program for Cycle 7". As indicated in our letter to the licensee dated September 14, 1978, we had concluded that the scope of the program was sufficiently comprehensive for assessment of current conditions of the SO-1 steam generators. The licensee implemented this program and provided the results of the inspections by letter dated October 25, 1978. Based on these results the licensee stated that tube denting and tube support plate degradation in the SO-1 steam generators have been arrested. Furthermore, the licensee stated that previous anti-vibration bar (AVB) wear problems have been eliminated by the new design of AVBs installed in October 1976.

By application dated July 31, 1978 (Proposed Change No. 75) the licensee requested that the Provisional Operating License for SO-1 be amended to add requirements to the Facility Technical Specifications for "Inservice Inspection of Steam Generator Tubes" and for limiting primary to secondary leakage through the steam generator tubes. The application was supplemented by letter dated October 24, 1978, which provided additional information.

Evaluation

Implementation of Steam Generator Inspection Program

The licensee's inspection program consisted of general surveillance, AVB area inspection, tube denting inspection, and photographic and videotape inspection in all three steam generators.

The general surveillance program consisted of eddy current testing (ECT) of tubes located at the intersection of every fourth row and column beginning with row six on the hot and cold leg sides in all three steam generators. Tubes on the hot leg side were inspected through the U-bend to just above the top support plate on the cold leg side. Tubes which could not be inspected through the U-bend from the hot leg side because of denting were inspected through the U-bend from the cold leg side if not restricted. Tubes on the cold leg side were inspected through the intersection of the first tube support plate where previous inspection had indicated tube imperfections. All tubes with ECT indications greater than 50% wall thinning were plugged.

As a result of the general surveillance program, 16.6% of the hot leg tubes and 12% of the cold leg tubes were inspected in steam generator A. In steam generator B, 18.1% of the hot leg tubes and 5.5% of the cold leg tubes were inspected and in steam generator C, 13.8% of the hot leg tubes and 5.9% of the cold leg tubes were inspected. The results of these inspections are summarized in Table I. Only two defective tubes were found. Tube R37-C49 in steam generator A had a 50% defect (wall thinning) at the tubesheet and tube R35-C47 in steam generator C had a 56% defect at the tubesheet. A total of 10 tubes with previous indications greater than 20% showed an increase in defect depth of greater than 10% since the last inspection and 58 tubes had new indication exceeding 20%.

TABLE I: SUMMARY OF GENERAL INSPECTION RESULTS

	STEAM GENERATOR A		STEAM GENERATOR B		STEAM GENERATOR C	
	INLET	OUTLET	INLET	OUTLET	INLET	OUTLET
NO. OF TUBES INSPECTED	630	458	685	226	480	226
NEW INDICATIONS >20%	0	24	4	5	24	1
PREVIOUS INDICATIONS >20% WITH >10% INCREASE FROM PREVIOUS INSPECTION	3	3	0	0	4	0
INDICATIONS >50% WITH >10% INCREASE FROM PREVIOUS INSPECTION	0	0	0	0	0	0
DEFECTIVE TUBES >50%	1	0	0	0	1	0

The AVB area inspection plan included all ECT of all tubes in steam generators A, B, and C which had ECT indications at the AVB intersection(s) during the October 1976 or September 1977 inspections. In addition, all tubes in rows 30 through 36, columns 12 through 17 and columns 84 through 89 where the tubes are in contact with the old AVBs but are not in contact with the new AVBs were inspected. Some tubes in the general inspection plan also contact the AVBs. These tubes were ECT inspected from the inlet side through the U-bend such that all AVB locations were inspected in each tube. Tubes which could not be inspected from the hot leg side because of denting were inspected from the cold leg side.

The average depth of the AVB wear identified by ECT in the September 1977 inspection was 30.1% in steam generator A, 32.7% in steam generator B, and 33.4% in steam generator C. The corresponding values in this inspection were 30.0% in steam generator A, 33.4% in steam generator B, and 32.2% in steam generator C. The difference in the mean values are within the error of ECT and indicate essentially no wear during the past year of operation.

The denting inspection program included gauging in all three steam generators of all tubes noted to be restricted during the April 1978 denting inspection, the tubes immediately adjacent to these tubes, and the tubes immediately adjacent to any tubes with newly discovered restrictions. These tubes were probed from the hot leg side through the top support plate. Using probe diameters of 0.560", 0.500", 0.460", and 0.400", restricted tubes were probed with sequentially smaller diameter probes until the passing diameter was determined and the dent magnitude was quanified within the range of probe diameters. Eddy current data obtained during the tube gauging program was analyzed to detect the presence of dents at the third and fourth support plates in steam generators A and C and all support plates in steam generator B.

Of the 332 tubes gauged in steam generator A, only one tube indicated an increase in dent magnitude and three tubes indicated decreases in dent magnitudes. These anamolies are believed to be within the accuracy of the inspection methods. Differences in equipment rigidity, probe wear, and equipment alignment can cause some variations in the results from different inspections. Forty-four tubes were gauged in steam generator B. All dented tubes found during this inspection were dented to the same magnitude in the April 1978 inspection. In steam generator C, 463 tubes were gauged. One tube on the outlet side, which had not been inspected previously restricted passage of a 0.460" probe. This tube was plugged. Five tubes that allowed passage of a 0.560 probe in the April 1978 inspection would not allow passage of that probe during this inspection; however, 43 tubes that were noted to restrict the 0.560" probe in the April 1978 inspection passed the 0.560" probe during this inspection. Although it seems unlikely that a large number of tubes would be dented to a magnitude just between two probe diameters, the 0.560" probe is the standard probe size, which could be interpreted as indicating that all of these tubes have experienced only minor denting, limited to a small range of dent magnitudes. Furthermore, based on experience at other plants, if denting were progressing in the steam generators almost all of the tubes inspected would have been affected. Therefore, the above indications of changes in dent magnitude are a result of using different gauging equipment and are not considered an indication of progression of denting.

The results of the ECT dent inspection program showed that all dents in the third and fourth support plates in steam generators A and C and in all four support plates in steam generator B had been previously detected and no progression of denting was observed.

Photographs and/or videotapes of the tube support plate flow slots were taken in all three steam generators through the secondary side manway and nozzle inspection ports just above the tube sheets. The condition of the flow slots in the top support plate in steam generator C was examined by videotape fibroscopic gauging through the three inch drilled inspection port.

Comparison of the widths of the flow slots in the lower support plates of all three steam generators between September 1977 and October 1978 indicate, within the error of measurement, no change in flow slot widths. Videotape fibroscopic measurements of the flow slots in the upper support plate of steam generator C also indicated, within the error of measurement, no change in the "as manufactured" dimensions of the flow slots.

Evaluation by the licensee and the NRC staff of photographs of the tube support plates in all three steam generators revealed no perceptible changes in the condition of the steam generators since the October 1977 inspection. The licensee did indicate the existence of a crack in the steam generator A support plate which had not been previously noted. The crack is located in the first support plate on the inlet side of the third flow slot from the manway near the center stay rod. It has been concluded by the licensee that the crack is not new, and we agree with this conclusion.

Photographs from the April 1978 inspection did not include the area of the support plate where this crack is located. Although a crack-like indication was observed at that location on the October 1977 inspection photograph, it was believed to be a shadow due to a sludge deposition. The licensee has reported that no detectable change in tha condition of the support plates were observed in the videotapes taken during this inspection.

Technical Specifications

The technical specifications proposed by the licensee include inspection requirements for random tube degradation, anti-vibration bar (AVB) tube wear, and tube denting. The inspection plan for random tube degradation conforms to the guidance in Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Denting and AVB wear are deterministic forms of tube degradation for which the mechanism and location of degradation are understood. Therefore, these types of degradation have been treated separately from other more random forms of tube degradation. Every inspection is required to include all the non-plugged tubes in one steam generator with imperfections at AVB intersections identified as greater than 30% in previous inspections. In the event that wear rates are exhibited which are in excess of the tube degradation allowance provided by the 50% plugging criteria, the situation will be reported to the Commission for approval of proposed remedial action. Each inspection is also required to include gauging of all previously identified restricted tubes in either steam generator A or C. If progression of denting is observed to be recurring it will be reported to the Commission for approval of the proposed remedial action.

We have concluded that the proposed Technical Specifications will adequately monitor the condition of the steam generators at San Onofre Unit 1. The general inspection program is based on the guidance in Regulatory Guide 1.83. The required inspections for denting and AVB wear were established after several discussions between representatives of the NRC and the licensee. We consider the methods established for monitoring the conditions of the steam generators with respect to these two phenomena to be acceptable.

The licensee's proposed primary to secondary leakage through the steam generator tubes is identical to the present license condition 3.F(2) which we have previously reviewed and found acceptable. Therefore, deletion of license condition 3.F(2) concurrent with its incorporation in the Technical Specifications is purely administative in nature.

We have concluded that the proposed Technical Specifications will assure maintenance of integrity of the steam generators for safe operation of SO-1. The results of the inservice inspections performed during the Cycle 7 refueling outage have confirmed that steam generator tube denting and support plate cracking have been arrested. We therefore found it acceptable to perform subsequent steam generator inservice inspections in accordance with the added Technical Specification 4.16 "Inservice Inspection of Steam Generator Tubing". Accordingly, license conditions 3.E and 3.F(1) concerning steam generator inspections may be deleted.

C. Technical Specifications Followup to Amendment No. 25

By letter dated April 1, 1977, we issued our Safety Evaluation (SE) supporting Amendment No. 25 to Provisional Operating License for SO-1. The amendment consisted of changes to the Technical Specifications required for operation of SO-1 with the Cycle 6 core, the modified ECCS features, the new sphere enclosure and the associated modifications in conjunction with a reduced exclusion area boundary.

In our April SE, we requested the licensee to propose before the refueling outage for Cycle 7 operation technical specifications requirements among other things: (1) inspection of the containment spray nozzles at five-year intervals, (2) periodic visual inspection of non-redundant containment spray piping, and (3) periodic verification that a minimum of 5400 lbs of trisodium phosphate is stored inside the containment. These requirements provide long-term assurance of the operability of safety-related systems assumed by us in evaluating the Loss-of-Coolant Accident (LOCA) with a reduced exclusion area boundary at SO-1.

By application dated March 31, 1978 (Proposed Change No. 70), the licensee requested an amendment to its Provisional Operating License. The amendment would modify the facility Technical Specifications by adding the above identified requirements, in response to the NRC staff positions in our April 1, 1977 SE.

The licensee proposed to specify the minimum allowable quantity of anhydrous trisodium phosphate to be stored in the containment sump, and the surveillance requirements to inspect the trisodium phosphate in the containment sump, the non-redundant portions of the containment spray system piping and the containment spray system nozzles.

To support the proposed Technical Specifications, the licensee stated in its letter dated September 30, 1978, that the dissolution of the minimum weight of 5400 lbs of anhydrous trisodium phosphate will ensure that the pH of the water in the sump will be greater than 7 within four hours after the LOCA. The licensee also stated that this pH will prevent chloride stress corrosion cracking of systems and components exposed to the sump water following a postulated LOCA. The proposed surveillance requirement for the trisodium phosphate is to assure that a sufficient inventory of the chemical is stored in the containment sump. The surveillance requirement for non-redundant piping in the containment spray system is to provide long-term assurance that the integrity of that system will be maintained.

Evaluation

Originally, in the August 8, 1977 application, the licensee proposed to inspect and test each containment spray nozzle at least once per refueling outage to verify no blockage. These nozzles are used to spray boric acid plus hydrazine into the containment during a LOCA. Periodic surveillance of the spray nozzles assures that the nozzles will perform the intended function in the event of a LOCA. During recent telephone discussions the licensee proposed instead to test each containment spray nozzle at least every second refueling outage to verify no blockage. The licensee stated that this modified frequency of testing the containment spray nozzles is the same as the frequency for similar testing for the fire protection spray and sprinkler system presently required in the Technical Specifications. The modified proposed frequency to perform the no-blockage test through the nozzles satisfies the NRC staff position in our April 1, 1977 SE. Therefore, the licensee's modified proposal for surveillance on the containment spray nozzles is acceptable.

Anhydrous trisodium phosphate is placed in the racks in the containment sump to raise the pH of the sump water during a LOCA to 7.0 or above. This value of pH is considered necessary to minimize the effects of chloride stress corrosion cracking on mechanical systems and components over the long term. The licensee proposed to provide a minimum of 5400 lbs of anhydrous trisodium phosphate in the containment. We have determined that this quantity of trisodium phosphate will ensure a minimum sump water pH of 7.0 within four hours following the design basis LOCA. This proposed technical specification satisfies the NRC staff position in our April 1, 1977 SE. Therefore, we conclude that the proposed added provision to Section 3.3.1 is acceptable.

The licensee also proposed surveillance requirements to ensure that a sufficient inventory of the trisodium phosphate is in the containment sump. We have reviewed the licensee's proposed change to

Sections 4.2.I.A(3) and 4.2.I.A(4) of the Technical Specifications. The licensee proposed to visually inspect each rack to verify that the racks are full, and have maintained their integrity. The licensee also proposed to conduct a test of the ability of an undisturbed sample of trisodium phosphate, from one of the separate sample storage racks, to raise the pH of borated water to at least 7 within four hours. The sample of the chemical would have been exposed to the same environmental conditions as in the main racks which contain the 5400 lbs trisodium phosphate. The test conditions would be similar to the conditions predicted for the design basis LOCA.

Based on our review and experience with trisodium phosphate stored in racks in containment sumps at other nuclear power plants, we conclude that the proposed Technical Specifications 4.2.I.A(3) and 4.2.I.A(4) are acceptable. However, it was necessary to more precisely specify the amounts of trisodium phosphate and the condition of the borated water for the test to determine the ability of the trisodium phosphate to raise the pH of the water expected in the sump during the design basis LOCA to a value of at least 7 within four hours from the start of the test. We therefore modified the licensee's proposed Technical Specification 4.2.I.A(3)(c) to conform to the acceptable provisions in the Westinghouse Standard Plant Technical Specifications for testing this chemical. The licensee has agreed to the modified wording.

We have concluded that proposed Specifications 4.2.I.A(3) and 4.2.I.A(4), as modified, provide assurance that there is sufficient inventory of trisodium phosphate in the containment sump and that this inventory will increase the value of pH of the water in the sump to at least 7 during the design basis LOCA within 4 hours. We have also concluded that the test conditions specified in proposed Specification 4.2.I.A(3)(c), as modified by us, are representative of the conditions expected in the sump water during the postulated LOCA. We therefore find that these proposed technical specifications as modified by us are acceptable.

The licensee proposed to visually inspect the non-redundant containment spray piping at intervals not to exceed the normal plant refueling intervals. The licensee is already required to perform inspections of the Class II containment spray system piping in accordance with the ASME Code requirements. The proposed visual inspections would augment the Code inspection requirements. Further, it is consistent with similar surveillance requirements previously accepted by us in Specification 4.2.II.C for verifying leaktight integrity of ESF equipment outside containment and in Specification 4.15.B(2)b. for fire protection involving the containment spray system piping. Based on the above, we conclude that the licensee's proposed interval to inspect the non-redundant containment spray system piping is acceptable. We also conclude that proposed technical specification 4.2.II.D meets the NRC staff position in our April 1, 1977 SE and is acceptable.

Environmental Conclusion

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 31, 1978

References

1. J. H. Drake, "SCE Amendment No. 75, Docket No. 50-206, Proposed Change No. 73 to the San Onofre 1 Technical Specification," Ltr., August 18, 1978.

2. KP Baskin, "Responses to NRC Questions on San Onofre Nuclear Generating Station, Unit 1, Cycle 7 Reload, September 1978, Westinghouse Proprietary Class 2," Ltr., September 22, 1978.

3. D. L. Ziemann, "Request for Additional Information," Ltr., Sept. 6, 1978.

4. D. H. Risher, et al, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis" WCAP-8963, June 1977.

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