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SUBJECT: Forwards formal response to NRC 851021 questionnaire on
Cycle 3 reload analysis rept. Fuel assembly misloading
analysis in support of Cycle 3 reload licensing also encl.

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December 5, 1985

Director, Office of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Branch Chief
Licensing Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

Enclosed for your information is a formal response to your questionnaire on the Cycle 3 Reload Analysis Report for San Onofre Nuclear Generating Station Unit 2 transmitted to SCE on October 21, 1985. In addition, SCE is providing you with a fuel assembly misloading analysis in support of Cycle 3 reload licensing.

If you have any questions regarding the enclosed information, please call me.

Very truly yours,

M O Medford / [signature]

Enclosures

cc: Harry Rood, NRC Project Manager
F. R. Huey, USNRC Senior Resident Inspector, Units 1, 2 and 3

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Question 1:

SONGS Technical Specification 3.4.1.2 allows operation in Mode 3 with only one reactor coolant loop and its associated steam generator and one associated reactor coolant pump. In addition, there is an increase in design bypass flow rate as well as in the number of assumed plugged steam generator tubes in Cycle 3. Have each of these items and the effect on core flow been accounted for in the uncontrolled CEA withdrawal event from a subcritical or low power condition? Explain why the initial core mass flow rate for this event in Cycle 3 is more than 15% higher than in the reference cycle.

Response:

Mode 3 Operation

SONGS 2 Technical Specifications do allow operation in Mode 3 with only one reactor coolant pump running. The CPC system would provide protection during a CEA bank withdrawal beginning in Mode 3. Since the CPC's may be bypassed at zero power, it would be possible to close the reactor trip breakers regardless of the trip status of the four CPC channels. Nevertheless, protection is provided by the Plant Protection System (PPS). The low RCS flow trip bypasses would be automatically removed when excore safety channel log power exceeds 10^{-4}%. In addition, the Hi log power trip setpoint is set at 0.89% and requires positive operator action between 10^{-4}% and 0.89% to remove the trip. If fewer than four reactor coolant pumps were operating, or if RCS temperature or pressure were outside the CPC wide range trip limits, a continuous reactor trip signal would be generated by all four CPC channels. An immediate reactor trip would terminate a CEA bank withdrawal event before significant power were generated. The Mode 3 event is less limiting than the CEA bank withdrawal event presented from Mode 2 initial conditions with all four pumps running which generated a high log power trip when power exceeded the analysis setpoint of 2% of full power.

RCS Coolant Flow Rate

The increased design bypass flow rate was used in the DNBR calculations for the uncontrolled CEA withdrawal event from subcritical or low power conditions.

The number of plugged steam generator tubes assumed in the analyses of all transients has increased for Cycle 3. Plugging additional steam generator tubes would decrease the nominal RCS coolant flow rate. However, the nominal flow rate is expected to remain above the minimum flow rate required by the Technical Specifications. The Cycle 3 analyses considered the entire range of operating conditions allowed by the Technical Specifications and thus implicitly includes the effect of the increased number of plugged steam generator tubes on RCS flow.

The flow rate used for Cycle 1 analysis corresponded to a volumetric flow which is lower than the minimum required by the Technical Specifications. This was unnecessarily conservative. The flow rate used for the Cycle 3, although higher than used for Cycle 1, is conservatively low.

It should be noted that the flow rate described in Table 7.4.1-3 as initial core mass flow rate is actually the total system flow corresponding to the minimum volumetric flow rate given in the Technical Specifications. The DNBR calculations correctly used this flow rate decreased by the assumed 3% bypass flow to obtain core flow.

Question 2:

Why is a 15% Doppler coefficient multiplier used in some events such as the steam line break whereas a 25% multiplier is used in others such as the increased main steam flow event?

Response:

The Doppler coefficient multiplier to account for calculational uncertainty assumed for Cycle 3 is 15%, as for previous cycles. For analyses for which the Doppler coefficient is not a critical input, an additional 10% was applied in order to bound variations in Doppler coefficients from cycle to cycle and thus avoid future reanalysis.

Question 3:

The uncontrolled CEA withdrawal event from subcritical conditions resulted in a peak linear heat generation rate in excess of the steady state acceptable fuel to centerline melt limit of 21 kw/ft. Describe the method used to calculate the maximum transient fuel centerline temperature.

Response:

The uncontrolled CEA withdrawal from subcritical predicts a Peak Linear Heat Generation Rate (PLHGR) in excess of 21 kw/ft, the steady state center line melt limit. The predicted PLHGR is a product of the Core Average Linear Heat Rate at full power (5.6), the peak fractional power reached during the transient (65%) and the maximum predicted 3-D peak (7.0). For the case of interest:

$$PLHGR = 5.6 \text{ kw/ft} * 0.65 * 7.0 = 25.5$$

Because this transient value of PLHGR is higher than the steady state limit, an assessment of the resultant fuel centerline temperature was performed.

The maximum centerline enthalpy of the fuel is defined as the initial enthalpy plus that added during the transient. The calculation assumed that no heat is transferred away from the centerline during the transient (i.e., adiabatic conditions). The deposited enthalpy was calculated as follows:

$$E_{DEPOSITED} = \frac{\text{Core Power x Time at Power x 3-D Peak}}{\text{Mass of } UO_2 \text{ Present}}$$

The core average energy deposition during the transient is less than one full power second. The mass of UO_2 is greater than 95.4×10^6gm for Cycle 3 and later cycles. The 3-D peak is 7.0.

Substituting and using appropriate conversion factors:

$$E_{DEPOSITED} = \frac{3410 \text{ MW-sec} \times 7}{95.4 \times 10^6 \text{gm}} \times \frac{239 \text{ Cal}}{\text{kw-sec}} \times \frac{10^3 \text{Kw}}{\text{MW}}$$
$$= 59.8 \text{ cal/gm}$$

The initial enthalpy was determined based on the initial temperature. The initial temperature is 540^oF which corresponds to an initial enthalpy of 16 cal/gm.

The total enthalpy is therefore 59.8 + 16 = 75.8 cal/gm. The temperature corresponding to this enthalpy is less than 2000^oF, which is well below the melting point of UO_2.

Question 4:

Does the CEA worth at trip (all rods out) given in Table 7.3.2-1 include a stuck CEA? Why is there a reduction in trip worth from the previous cycle?

Response:

The CEA worth at trip given in Table 7.3.2-1 does include a stuck CEA. There is a reduction in trip worth from the previous cycle in order to bound variations in CEA trip worth in future cycles and so avoid future reanalysis.

Question 5/Question 6:

5. The staff has generically approved the use of a DNB statistical convolution technique for determining fuel rod failures only for the seized shaft loss of flow event as presented in CENPD-183. The use of this convolution technique for any other accidents is a deviation from our current practice, as described in the Standard Review Plan, where fuel damage is assumed whenever the DNBR is calculated to fall below the minimum DNBR criterion. This DNBR limit is defined such that there is a 95% probability with a 95% confidence level that a fuel rod will not experience DNB whenever the DNBR is above the limit. In the past, we have informed CE that the convolution technique is not approved for other accidents and is not presently under review. In view of this position, are there any events for Cycle 3 in which our radiological dose acceptance criteria would not be met using the CE-1 95/95 limit of 1.31 as the basis for fuel rod failure?

6. With reference to the preceding statement, please present the resulting offsite doses for the following reanalyzed Cycle 3 events so that we may confirm that they meet our acceptance criteria of being well within the guideline values of 10 CFR Part 100: (a) Increased main steam flow with loss of AC power, (b) Inside containment steam line break with loss of AC power.

Response:

The DNB Statistical Convolution Technique was used in the Cycle 3 Reload Analysis Report as the basis for determining fuel rod failure. Of the events presented, the following used the Statistical Convolution Technique:

(i) The Seized Shaft Loss-of-Flow Event,

(ii) Increased Main Steam Flow with Loss of AC, and

(iii) Inside Containment Steam Line Break with Loss of AC.

The NRC staff has generically approved the use of the DNB statistical convolution technique for Event (i), as presented in CENPD-183. Both Events (ii) and (iii), for which the NRC staff has not approved the use of the DNB statistical convolution technique for determining fuel rod failure, will be reanalyzed. Both cases can satisfy the radiological dose acceptance criteria using the CE-1 95/95 limit of 1.31 as the basis for fuel rod failure calculations. A brief discussion of these two cases is shown below.

Increased Main Steam Flow with Loss of AC

The radiological acceptance criterion for this event is that the radiological doses at the site boundary be bounded by the 10 CFR 100 dose limits.

The Increased Main Steam Flow event has been reanalyzed with modified initial conditions based on the actual DNBR margin expected to be present for Cycle 3 and beyond. This has the effect of initiating the transient at a larger DNBR value and hence the minimum DNBR attained during the transient is larger than in the original analysis.

As a result of this reanalysis, the results of this event are expected to show less than 5.0% failed fuel and less than 75 rem dose to the thyroid at the site boundary, using the 95/95 DNBR limit of 1.31 as the basis for fuel rod failure calculations. The details of this reanalysis will be submitted to the NRC staff in a later transmittal.

<u>Inside Containment Steam Line Break with Loss of AC</u>

The analysis of this event presented in the Cycle 3 Reload Analysis Report takes no credit for a CPC trip. The results (assuming the 95/95 DNBR limit of 1.31 as the basis for fuel rod failure) showed acceptable dose consequences (approximately 200 rem thyroid) associated with 18.2% fuel failure. The NRC staff has expressed some concern over the magnitude of the predicted fuel failures. In response, the event will be reanalyzed taking credit for a Variable Overpower Trip (VOPT) from the CPC's.

The VOPT uses input from the ex-core detector channels and the power is temperature compensated using input from the cold-leg RTD's. The qualification of these instruments was established in SCE's Environmental Qualification Report, with NCR's approval documented in NUREG-0712, Supplement No.4 (Safety Evaluation Report related to the operation of San Onofre Nuclear Generating Station, Units 2 and 3). The cited report establishes ex-core channel operability for 55 seconds and RTD operability for 30 minutes under conservatively specified accident conditions. This instrumentation qualification far exceeds the requirements for the subject steam line break accident.

In the event that fuel failure is calculated as a result of this transient, the appropriate acceptance criteria is that the radiological doses at the site boundary be bounded by the 10 CFR 100 dose limits.

The reanalysis of the Inside Containment Steam Line Break will show that the reactor trip will occur much earlier and, as a result, the calculated fuel failure associated with this event will be significantly reduced. The new analysis is expected to demonstrate that < 10% of the pins fail and that the site boundary dose to the thyroid is $\leq$ 75 rem. A further result of reanalyzing this event is expected to show that the radiological consequences of this event are less limiting than those of the Outside Containment Steam Line Break.

The Outside Containment Steam Line Break is being reanalyzed using the 95/95 DNB limit of 1.31 to calculate fuel rod failures. The results of this reanalysis will be presented to the NRC in a later transmittal as well.

<u>Summary</u>

1. Preliminary results show acceptable dose consequences for all events using NRC Staff's approved methods for calculating fuel rod failures based on the 95/95 DNBR limit of 1.31 except for Sheared Shaft which uses the DNB Statistical Convolution Technique.

2. The Increased Main Steam Flow with loss of AC event will be reanalyzed to credit the actual DNBR margin expected to be present in Cycle 3 and beyond. As a result, the dose consequences will be well within 10 CFR 100 limits.

3. The Inside Containment Steam Line Break event will be reanalyzed crediting the CPC VOPT. As a result, the dose consequences of this event are expected to be less limiting than the Outside Containment Steam Line Break which will be reanalyzed as the limiting radiological dose event in this category.

Question 7:

Measurement of CEA reactivity worth during Cycle 3 startup testing are proposed to be accomplished by the CEA exchange technique. The staff has not reviewed nor approved this rod swap technique for CE plants but has been informed by CE that a generic topical report would be submitted in the future. Based on this, the proposed use of the CEA exchange technique during startup testing is not acceptable at this time.

Response:

The proposed CEA exchange technique was employed to measure the CEA reactivity worth during the Cycle 2 startup on SONGS 2. These measurements were performed as part of a demonstration test performed during the SONGS 2 cycle startup. The demonstration test provided an acceptable result comparable with the measurements derived from boron/dilution technique with a substantial savings in critical path time and radwaste generation. SCE believes it is a viable and superior alternative for CEA worth measurements and shutdown margin verification without any compromise to safety requirements. SCE intends to apply this technique in subsequent startups under 10 CFR 50.59. Accordingly, a Safety Evaluation of the proposed CEA exchange technique as outlined in the Cycle 3 RAR will be prepared and test records will be maintained on file. The written Safety Evaluation which provides the bases for the proposed CEA exchange technique will then be made available to the NRC for information upon request.

Additionally, the CE Owner's Group plans to submit a topical report on the CEA exchange technique in December. This report concludes that the exchange technique has been demonstrated to provide data comparable to that using the boron/dilution method to verify the adequacy of control rod worths as predicted for the core design (based on the demonstration tests performed at SONGS and Palo Verde Nuclear Generation Station). It is concluded that the exchange technique may be generally applied for any cycle as an alternate to the boron/dilution method. This report will provide additional information to support SCE's position under 10 CFR 50.59 and should not be construed as a prerequisite to the intended application of this method on subsequent startup testing.

Supplementary Information on Fuel Assembly Misloading

The likelihood of an error in core loading is considered to be extremely remote because of the strict procedural controls employed. First, the core loading plan results in all assembly serial numbers facing the same direction making it virtually impossible to overlook a mis-oriented assembly. Second, a "tag board" is provided in the main control room showing schematic representations of the reactor core, spent fuel pool and new fuel storage areas. During core loading this tag board is used by a designated member of the reactor operations staff to constantly monitor the exact location of every fuel assembly. Finally, a visual core loading scan is performed and video recorded to verify that all assemblies have been correctly loaded before the reactor vessel head is secured.

If, in spite of the many precautions described above, it is postulated that a fuel assembly is misloaded, several situations may be possible. Comparisons of at least one CECOR measured power distribution in the 15-30% power range with predictions are included in the startup test procedures. These comparisons will detect most fuel assembly misloadings even with coincidental in-core failures in the neighborhood of the postulated misloading.

For a misload where two adjacent assemblies are interchanged, a difference in infinite multiplication factors of approximately 0.15 is detectable. For misloadings where distant assemblies are interchanged, the detectable difference is approximately 0.08.

Should a misloading occur which is not detectable at startup, two possibilities ensue: either (1) the misloading has little or no impact on margins to safety limits because there is little difference neutronically between the correct and incorrect loadings, or (2) the misloading is undetectable only near beginning of cycle but develops into a growing power distribution anomaly as the core depletes. This second case could be postulated for the SONGS-2 Cycle 3 design because of the loading of fresh assemblies with solid burnable absorbers. Examples of an initially undetectable fuel misloading were presented in the FSAR where it was demonstrated that these would become detectable before the onset of any fuel damage.

For the latter case, there is again little impact on the safety of the plant because the power distribution monitoring systems, COLSS and CECOR, will detect the anomaly before a Specified Acceptable Fuel Design Limit (SAFDL) is violated. In addition, continuous monitoring of in-core detector azimuthal tilt with COLSS will incorporate much of the power distribution anomaly directly into the calculation of the allowed core Power Operating Limit. Similarly, Technical Specification 3.2.3 requires that CPC maintain a conservative azmuthal tilt multiplier in its calculations of core conditions. If the azimuthal tilt grows to the 3-5% range, ordinary prudence suggests that the source of the anomaly be investigated. Should the tilt further increase to 10%, Technical Specification 3.2.3 requires that core power be reduced to 50% until the cause is identified and corrected.

Technical Specification 3.2.2 requires that a CECOR snapshot be taken at a minimum of every 31 Effective Full Power Days (EFPD's) to determine that the measured planar radial peaking factors obtained by using the in-core detector system are less than or equal to those installed in COLSS and CPC. Any unexpected increase in radial peaking factors would be immediately incorporated in COLSS and CPC calculations.

Since the maximum pin peak increase over any 31 EFPD interval would be determined by the maximum burnable poison depletion rate during that period, it is concluded that the maximum increase would be no worse than for the cases analyzed for the FSAR. As a result, it is concluded that the FSAR analysis remains bounding for the SONGS-2 Cycle 3 reload design.

SPW:5389F