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JUN 0 7 1977

Docket Nos. 50-361
and 50-362

Mr. Myer Bender
Chairman, Advisory Committee
on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Bender:

SUBJECT: REGULATORY STAFF SUPPORT OF ACRS ACTIVITIES ON SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

The NRC staff has agreed to assist the ACRS in the early identification of potential problem areas and potentially difficult novel features for each application. We have further agreed to advise you whether our acceptance reviews of tendered applications revealed any features of the application that cause a particular problem relative to any of the ACRS generic items. We have completed our acceptance review for San Onofre Units 2 and 3 Final Safety Analysis Report and provide the following information in response to these two agreements.

The design of San Onofre Units 2 and 3 is based upon prior Combustion Engineering experience with Arkansas Nuclear One Unit 2 (Docket No. 50-368) and Pilgrim Station Unit 2 (Docket No. 50-471). Differences in the NSSS for these designs are summarized in Enclosure 1.

We do not consider the differences in Enclosure 1 to be current problem areas or difficult novel features, nor do we consider it likely that these items will lead to technical problems in completing the review. One particular feature of the application possibly significant enough to warrant your interest is the Core Protection Calculator System (CPCS). This feature is similar to the ANO-2 design which the staff is reviewing. The applicant has had repeated meetings with NRR to describe his CPCS design as it has progressed. The design is to be translated into software

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and hardware and tested in the vendor's laboratory and the test results submitted for NRC approval. We expect that the basic CPCS design will be approved on the ANO-2 docket; modification to the design for San Onofre application will be reviewed by the staff.

Another significant feature of the San Onofre 2 and 3 plant is the unusually high seismic design criteria. The design SSE ground acceleration for the facility is 0.67g. We expect to perform an especially detailed and thorough review of the capability of seismic Category I structures and components to withstand this severe natural phenomenon. We also plan to thoroughly investigate all the available seismological and geological data to verify the adequacy of the design g value for the San Onofre site.

Our acceptance review for San Onofre 2 and 3 included a check to see if any features of the application cause any of the ACRS generic items identified in Mr. Bender's letter of February 24, 1977 to be a particular problem. We concluded that while several of these items will be applicable and will be included in our review, no features of this design cause any of the generic items to be a particular problem at this early review stage. The Group II (unresolved) generic items which we believe to be applicable to San Onofre 2 and 3 are identified in Enclosure 2. Our Safety Evaluation Report for San Onofre 2 and 3 will contain an appendix addressing unresolved ACRS generic items relative to this application.

Sincerely,

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosures:
1. Summary of Design Differences
2. Applicability of ACRS Generic Items to San Onofre 2 & 3

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SURNAME	MMMlynczak:mt	HRood	KKniel	DBVassallo		
DATE	6/2/77	6/6/77	6/6/77	6/6/77		

ENCLOSURE 1

SUMMARY OF NSSS DESIGN DIFFERENCES FOR SAN ONOFRE UNITS 2 AND 3 AS COMPARED TO ARKANSAS NUCLEAR ONE, UNIT 2 AND PILGRIM STATION, UNIT 2

Item	San Onofre Units 2 and 3	PSAR Reference Section	ANO-2	Pilgrim Station Unit 2
Hydraulic and Thermal Design Parameters				
Rated core heat output, MWt	3,390	4.4	2,815	3,456
Hot channel factors,				
Heat flux, Fq	2.35		2.35	2.35
Enthalpy rise, F_H	1.55	4.4	1.55	1.72
DNB ratio at nominal conditions	2.07 (CE-1)	4.4	2.26 (W-3)	2.26 (W-3)
Coolant flow				
Total flowrate, lb/h	148×10^6	4.4	120.4×10^6	148×10^6
Heat transfer at 100% power				
Average heat flux, Btu/h-ft^2	182,400	4.4	182,200	184,000
Maximum heat flux, Btu/h-ft^2	428,000	4.4	425,800	429,900
Maximum thermal output, kW/ft	12.5	4.4	12.5	12.6
Core Mechanical Design Parameters				
Number of fuel rods	49,500	4.2	40,716	49,476
Number of control assemblies, full/part-length	83/8	4.2	73/8	83/8

Item	San Onofre Units 2 and 3	PSAR Reference Section	ANO-2	Pilgrim Station Unit 2
Nuclear Design Data				
Structural characteristics				
Core diameter, in. (equivalent)	136	4.2	123	136
Core height, in. (active fuel)	150	4.2	150	150
Number of fuel assemblies	217	4.2	177	217

ENCLOSURE 2

APPLICABILITY OF ACRS GENERIC ITEMS TO SAN ONOFRE UNITS 2 AND 3

The Advisory Committee on Reactor Safeguards periodically issues a report listing various matters of generic concern to large light-water reactors. The most recent such report was issued on February 24, 1977. The generic matters are divided into Group I (Resolved) and Group II (Resolution Pending).

The listing that follows identifies the Group II generic concerns that we believe are applicable to San Onofre Units 2 and 3. These will be addressed in the Safety Evaluation Report to be issued for San Onofre Units 2 and 3.

II-1.	Turbine Missiles	Applicable.
II-2.	Operation of Containment Spray in LOCA	Applicable.
II-3.	Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock	Applicable. RGs 1.2 & 1.99 apply. (Sections 5.2 and 5.3).
II-4.	Instruments to Detect Fuel Failures	Applicable.
II-5.	Monitoring for Excessive Vibration or Loose Parts Inside Pressure Vessel	Applicable. Pending RG "Loose Parts Monitoring Program for the Primary System" will be used. (Section 5.2.8.6).
II-6.	Non-Random Multiple Failures	Applicable. WASH-1270 and Staff's December 1975 report applies.
II-7.	Behavior of Reactor Fuel Under Abnormal Conditions	Applicable (Section 4.4.3).
II-8.	BWR Recirculation Pump Overspeed During a LOCA	Not applicable to PWRs.
II-9.	Advisability of Seismic Scram	Applicable. No trip required pending generic determination by staff.

II-10.	Emergency Core Cooling System Capability for Future Plants	Not applicable to plants undergoing OL review.
IIA-1.	Control Rod Drop Accident (BWRs)	Not applicable to PWRs.
IIA-2.	Ice Condenser Containments	Not applicable to dry containments. (Tables 15.4-3 and 15.4-3a)
IIA-3.	Rupture of High Pressure Lines Outside Containment	Applicable. (Section 3.6)
IIA-4.	PWR Pump Overspeed During a LOCA	Applicable.
IIA-5.	Isolating Low Pressure Systems Connected to RCSB	Applicable. (Section 5.4.7)
IIA-6.	Steam Generator Tube Failures	Applicable. (Section 5.4.2)
IIA-7.	Periodic Comprehensive 10-year Review of Operating Power Reactor	Not applicable to non-operating plants.
IIB-1.	Computer Reactor Protection System	Applicable. (Section 7.2)
IIB-2.	Qualification of New Fuel Geometries	Applicable. (Section 1.5)
IIB-3.	Behavior of BWR Mark II Containments	Not applicable to PWRs.
IIB-4.	Stress Corrosion Cracking in BWR Piping	Not applicable to PWRs.
IIC-1.	Locking Out of ECCS Power-Operated Values	Applicable. (Section 6.3)
IIC-2.	Design Features to Control Sabotage	Applicable. (Section 13.6)
IIC-3.	Decontamination and Decommissioning of Reactors	Applicable. Results of AIF study will be considered.

IIC-4.	Reactor Vessel Supports (Asymmetric LOCA Loads from sudden Subcooled Blowdown)	Applicable. Analytical models, dynamic analyses and interface requirements will be reviewed. (Section 3.6, 5.4)
IIC-5.	Water Hammer	Applicable. (Section 10.4)
IIC-6.	Maintenance and Inspection of Plants	Applicable. (Section 12)
IIC-7.	Behavior of BWR Mark I Containments	Not applicable to PWRs.
IID-1.	Safety-Related Interfaces Between Reactor Island and Balance of Plant	Not applicable.
IID-2.	Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Applicable.