RELOAD SAFETY EVALUATION

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 1, CYCLE 8

January, 1980

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

The San Onofre Nuclear Generating Station Unit 1 is in its seventh cycle of operation. The unit will refuel and be ready for Cycle 8 startup in May 1980.

This report presents an evaluation for Cycle 8 operation which demonstrates that the core reload will not adversely affect the safety of the plant. It is not the purpose of this report to present a reanalysis of all potential incidents. Those incidents analyzed and reported in the FSA[1] which could potentially be affected by fuel reload have been reviewed for Cycle 8 design described herein. The results of new analyses have been included, and the justification for the applicability of previous results from the remaining analyses is presented. These analyses assume that: (1) Cycle 7 operation is terminated between 10030 and 11030 MWD/MTU, (2) Cycle 8 burnup is limited to the end-of-full power capability*, and (3) there is adherence to plant operating limitations given in the technical specifications, and (4) the proposed technical specification basis change in Section 4 is implemented.

The San Onofre 1, Cycle 8 core loading pattern is shown in Figure 1. The one Region 6 and 51 Region 7 fuel assemblies from Cycle 7 will be removed and replaced by 52 Region 10 fuel assemblies. A Region 7 fuel assembly will be reused in the central core position.

Nominal design parameters for Cycle 8 are 1347 Mwt core power, 2100 psia system pressure, nominal core inlet temperature of 553°F, and 4.64 kw/ft average linear fuel power density.

*Definition: Full rated power and temperature (approximately 575°F T_{avg}), control rods fully withdrawn, and zero ppm of residual boron.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the Region 10 fuel assemblies is the same as the Region 9 assemblies. Table 1 compares pertinent design parameters of the various fuel regions. The Region 10 fuel has been designed according to the fuel performance model in Reference 2.

Clad flattening will not occur during Cycle 8. All fuel regions have a predicted clad flattening time equal to or greater than 50,000 EFPH. No fuel region will receive this exposure.

2.2 NUCLEAR DESIGN

Cycle 8 core loading statisfies an ECCS analysis limit of $F_Q^T \times P \leq 2.95$[3][4] as shown in Figure 2. The limitations on F_Q^T of 2.95 include the effects of the local power peaking of Figure 3.1 in WCAP 8131[5] to assure that the allowable value for LOCA is satisfied. The points plotted on Figure 2 include maneuvers typically done at San Onofre Unit 1 and variants on these maneuvers done at a number of control rod insertions, times and burnups.

The limiting F_Q^T has been determined for the combination of the most adverse F_{XY} and the most adverse F_Z^N that will be experienced during operation in Cycle 8. The most adverse F_{xy} occurs at beginning of life and the most adverse F_Z^N occurs at end of life. The results shown for F_Q^T in Figure 2 include uncertainty factors of 15% for conservatism and 4% for manufacturing tolerances.

The xenon transient analysis has been evaluated similarly to analyses of previous cycles. The most limiting F_Z^N, including an uncertainty of 10% on F_Z^N is 1.87 at 84% of core height. With the Cycle 8 $F_{\Delta H}^N$ of 1.55, an F_Z^N of 1.96 would be required to reach a DNBR of 1.30 at this elevation and 118% power. This margin exists assuming a control rod withdrawal occurs with the rods moving to the fully withdrawn position.

Table 2 provides a comparison of Cycle 8 kinetics characteristics with the current limit based on previously submitted accident analysis. The effect of the Table 2 parameters, including those that fall outside the current limits, are evaluated in Section 3. Table 3 provides the end-of-life control rod worths and requirements at the most limiting condition during the cycle. The required shutdown margin is based on a previously submitted accident analysis.(3)(6) The available shutdown margin exceeds the minimum required to meet the accident analysis.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins will result from the Cycle 8 reload. The present DNB core limits have been found to be conservative. The evaluation utilized the Cycle 5 F_Z^N tradeoff curve.(7) This curve remains conservative for Cycle 8. In conducting the F_Z^N tradeoff evaluation, the local power spike due to fuel densification was not included as justified in Reference 8.

3.0 ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability is evaluated considering the consequences of those incidents examined in the FSA,[1] using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at 100% of rated power during Cycle 8. For Condition II overpower transients, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 8 core. The time dependent densification model[8] was used for fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining F_Q^T at or below 2.95. This limit is satisfied by the power control maneuvers allowed by the technical specifications, which assure that the Interim Acceptance Criteria (IAC) limits are met for a spectrum of small and large breaks.

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSA[1] were examined. In most cases, it was found that the effects were accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis. For those incidents which were reanalyzed, it was determined that the applicable design bases are not exceeded, and, therefore, the conclusions presented in the FSA are still valid.

A core reload can typically affect accident input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 8 parameters in each of these three areas were examined as discussed below to ascertain whether new accident analyses were required.

A comparison of Cycle 8 core physics parameters with current limits is given in Table 2. The kinetic values fall within the bounds of the current limits.

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths, and trip reactivity. Tables 2 and 3 show that the maximum reactivity withdrawal rate, and the shutdown margin with the worst stuck RCCA are within the current limits. The ejected rod worths and trip reactivity curve are within the bounds of the previous Cycle 7 evaluation.

Peaking factor evaluations were performed for the rod out of position, dropped RCCA bank, dropped RCCA, and hypothetical steamline break accident to ensure that the minimum DNB ration remains above 1.30. These evaluations were performed utilizing Cycle 8 transient statepoint information and peaking factors. In each case, it was found that the peaking factor for Cycle 8 was lower than the value for which DNBR equals 1.30. Consequently, no further investigation or analysis was required. The peaking factors following control rod ejection are within the limits of previous analysis for the EOL zero power and full power cases. Peaking factors for the Cycle 8 BOL zero and full power incidents exceed previously analyzed values, and these cases are reanlayzed in Section 3.3.

3.3 INCIDENTS REANALYZED

The control rod ejection analysis is affected adversely by increased peaking factors following rod ejection for the beginning-of-life hot full power (HFP) and hot zero power (HZP) cases. The two cases shown in Table 4 were reanalyzed, and as shown in Table 5, the hot spot fuel rod does not exceed the limiting fuel criteria[9] for either case.

4.0 TECHNICAL SPECIFICATIONS

This section contains the technical content of proposed changes to the Technical Specifications. The purpose of the change is to make the Technical Specifications consistent with the rod ejection accident safety criteria while removing cycle dependent values from the Technical Specifications.

3.5.2 Control Insertion Limits (Basis)

The current item 3 of the Basis on p. 3-25 of the Technical Specifications is as follows:

3. The maximum ejected rod worth is limited to 0.21% p at HFP-BOL, 0.68% p at HZP-BOL, 0.15% p at HFP-EOL, and 0.58% p at HZP-EOL. The resulting maximum fuel centerline temperatures are 4789°F, 44
1826°F, 4719°F, and 1231°F, respectively. The rod insertion 10/31/78
limits restrict ejected rod worths to less than the above values.

This item of the Basis is violated for the Cycle 8 RSE. In order to have acceptable rod ejection results dependent only on its accident safety criteria, the current item 3 Basis should be replaced by the following proposed revised Basis:

"3. The worst-case ejected rod accident covering HFP-BOL, HZP-BOL, HFP-EOL and HZP-EOL shall satisfy the following accident safety criteria[8]:

a. Average fuel pellet enthalpy at the hot spot is below 225 cal/gm for non-irradiated fuel and 200 cal/gm for irradiated fuel.

b. Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot."

Add to the references of Section 3.5 of the Technical Specifications the following:

"8. An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January 1975."

5.0 REFERENCES

1. Docket Number 50-206, "San Onofre Nuclear Generating Station, Unit 1, Part 2, Final Safety Analysis".

2. Miller, J. V. (Ed.), "Improved Analytical Model Used in Westinghouse Fuel Rod Design Computations", WCAP-8785, October 1976.

3. SCE Report, "ECCS Performance Reanalysis, San Onofre Nuclear Generating Station Unit 1, May 1978", Submitted by letter dated May 30, 1978, Docket No. 50-206, K. P. Baskin to D. Eisenhut.

4. Amendment No. 35 to San Onofre Unit 1 Provisional Operating License DPR-13; Subject: Revision to Appendix A Technical Specifcations; Docket No. 50-206, August 4, 1978.

5. "Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station Unit 1, Cycle", WCAP-8381, May 1973.

6. SCE Report, "Steamline Break Accident Reanalysis, San Onofre Nuclear Generating Station Unit 1, October 1976", Attachment to letter, K. P. Baskin to K. R. Goller, December 29, 1976.

7. "Description and Safety Analysis Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5", Attachment to letter from Jack B. Moore to Edson G. Case, March 7, 1975.

8. Hellman, J. M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation", WCAP-8218-P-A, March 1975 (Proprietary) and WCAP-8219-A, March 1975 (Non-Proprietary).

9. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP 7588, Revision 1-A, January 1975.

10. Skaritka, J., Editor, "Reload Safety Evaluation - San Onofre Unit 1, Cycle 7", August 1978.

TABLE 1

SAN ONOFRE UNIT 1 - CYCLE 8

Fuel Assembly Design Parameters

Region	7	8	9	10
Enrichment (w/o U-235), Nominal	4.00	3.99	3.98	3.98
Density (% Theoretical)*	94.65	94.59	94.66	94.66
Number of Assemblies	1	52	52	52
Approximate Burnup at Beginning of Cycle 8 (MWD/MTU)	29050	20800	8750	0

*As built values - all fuel regions

TABLE 2

SAN ONOFRE UNIT 1 - CYCLE 8

Core Physics Parameters

	Current Limit	Cycle 8
Moderator Temperature Coefficient, ($\Delta\rho/^{o}F$) x 10^4	-4.0 to 0(1)	-3.3 to -0.3
Doppler Coefficient, ($\Delta\rho/^{o}F$) x 10^5	-2.75 to -1.4(10)	-2.6 to -1.4
Delayed Neutron Fraction, β_{eff}, (%)	0.50 to 0.70(1)	0.55 to 0.62
Maximum Prompt Neutron Lifetime (μ sec)	26(7)	11.4
Maximum Reactivity Withdrawal Rate, (pcm/sec)*	40(10)	<40

*pcm $\equiv 10^{-5}\ \Delta\rho$

TABLE 3

SAN-ONOFRE UNIT 1 - CYCLES 7 and 8

Shutdown Requirements and Margins

	Cycle 7		Cycle 8	
	BOL	EOL	BOL	EOL
Control Rod Worth (% $\Delta\rho$)				
All Rods Inserted	6.7	7.3	6.8	7.4
All Rods Inserted Less Worst Stuck Rod	5.5	6.2	5.9	6.4
(1) Less 10%	4.9	5.6	5.4	5.8
Control Rod Requirements (% $\Delta\rho$)				
Reactivity Defects (Doppler, Tavg, Void, Redistribution	1.9	2.6	2.0	2.7
Rod Insertion Allowance	0.8	0.8	0.9	0.9
(2) Total Requirements	2.7	3.4	2.9	3.6
Shutdown Margin [(1)-(2)](% $\Delta\rho$)	2.2	2.2	2.5	2.2
Required Shutdown Margin (% $\Delta\rho$)	1.25	1.9[(3)(6)]	1.25	1.9

TABLE 4

SAN ONOFRE UNIT 1 - CYCLE 8

Rod Ejection Parameters

	Previous Analysis Values(10)	Value* Used In Reanalysis
HZP - BOL		
Max. Ejected Rod Worth, % $\Delta\rho$	0.68	0.68
Max. F_Q	8.47	8.95
β_{eff}	0.0055	0.0055
HFP - BOL		
Max. Ejected Rod Worth, % $\Delta\rho$	0.21	0.21
Max. F_Q	5.48	6.11
β_{eff}	0.0055	0.0055

HZP - Hot Zero Power

BOL - Beginning of Life

HFP - Hot Full Power

*These values bound the Cycle 8 values

TABLE 5

SAN ONOFRE UNIT 1 - CYCLE 8

Results of Rod Ejection Analysis - Hot Spot Fuel and Clad Temperatures

	BOL	BOL
Initial Power, %	103%	0%
Maximum Fuel Pellet Center Temperature (^{o}F)	4892	2863
Maximum Fuel Average Temperature (^{o}F)	3774	2446
Maximum Clad Average Temperature (^{o}F)	2290	1754
Maximum Fuel Enthalpy (cal./gm)	162.7	97.7

FIGURE 1

CORE LOADING PATTERN

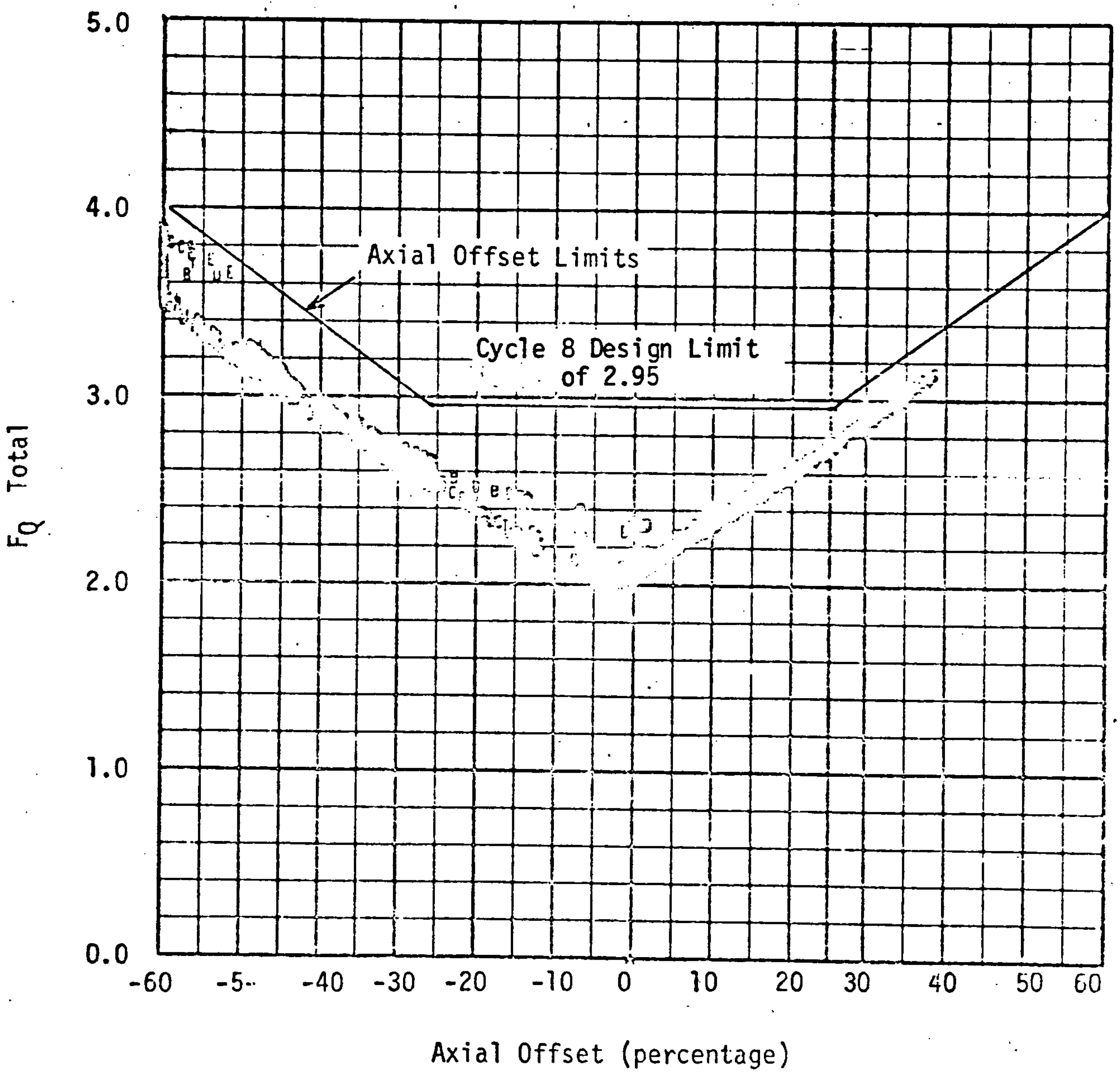
SAN ONOFRE UNIT 1 CYCLE 8

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A							8	10	8						
B					10	10	10	9	10	10	10				
C				10	10	9	9	8	9	9	10	10			
D			10	10	8	9	8	10	8	9	8	10	10		
E		10	10	8	8	9	8	8	8	9	8	8	10	10	
F		10	9	9	9	8	9	8	9	8	9	9	9	10	
G	8	10	9	8	8	9	9	9	9	9	8	S 8	9	10	8
H	10	9	8	10	8	8	9	7	9	8	8	10	8	9	10
J	8	10	9	S 8	8	9	9	9	9	9	8	8	9	10	8
K		10	9	9	9	8	9	8	9	8	9	9	9	10	
L		10	10	8	8	9	8	8	8	9	8	8	10	10	
M			10	10	8	9	8	10	8	9	8	10	10		
N				10	10	9	9	8	9	9	10	10			
P					10	10	10	9	10	10	10				
R							8	10	8						

S	Source Location in Fuel
X	Region Number

Figure 2

F_Q Total vs. Axial Offset for San Onofre
Unit 1 - Cycle 8



DESCRIPTION OF PROPOSED CHANGE AND
SAFETY ANALYSIS
PROPOSED CHANGE NO. 90 TO THE
TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE DPR-13

This is a request to (1) revise the definition of Containment Integrity as contained in Section 1.0; (2) add requirements for calibration and testing of Auxiliary Feedwater Flow and Condensate Tank level instrumentation in Table 4.1.1 of Section 4.1, and; (3) revise the requirements for the containment isolation system design in Section 5.2 of the Appendix A, Technical Specifications for San Onofre Unit 1.

Reason for Proposed Change

These changes are submitted to provide revisions and additions to reflect station modifications being performed during the January, 1980 outage to complete the Category A NRC TMI Short Term Lessons Learned Requirements as described in our letters dated January 17 and 23, 1980.

Existing Specifications

The existing specifications are as constituted in Sections 1, 4, and 5 of the Appendix A, Technical Specifications for Provisional Operating License DPR-13.

Proposed Specifications

The definition of "Containment Integrity" in Technical Specifiation 1.0 would be revised to read:

"Containment Integrity":

Containment Integrity means that all of the conditions below are satisfied:

(1) All manual containment isolation valves (or blind flanges) are closed.

(2) The equipment door is properly closed.

(3) At least one door in each personnel air lock is properly closed.

(4) All automatic and remote manual containment isolation valves are operable."

Technical Specification 4.1 would be revised by adding items 20 and 21 to Table 4.1.1 to read:

	Channels	Action	Minimum Frequency
"20.	Auxiliary Feedwater flow	Test	Once per month during operation

21.	Condensate Tank level	Calibration	At each refueling shutdown
		Test	Once per month during operation"

The last paragraph of Technical Specification 5.2 would be revised to read:

"The automatically actuated containment isolation valves shall be designed to close upon high-pressure in the containment (set point no higher than 5 psig) or upon safety injection initiation. In addition, design provisions shall prevent automatic reopening of any isolation valves upon reset of the containment isolation signal. The actuation system shall be designed such that no single component failure will prevent containment isolation if required."

<u>Safety Analysis</u>

Each of the proposed Technical Specification revisions discussed above is required as part of the implementation of the NRC TMI Short Term Lessons Learned Requirements. The basis for each revision is discussed below:

1. By letters dated December 17, 1979 and January 17, 1980, the results of the Essential/Non-essential study of the containment isolation systems were provided to the NRC as required by NUREG-0578, Section 2.1.4. Based on the results, two automatically isolated systems which were previously identified as non-essential systems have been identified as essential systems (i.e., those required to mitigate an accident or which, if unavailable, could increase the magnitude of the event). These systems are:

   a. The turbine plant cooling water supply and return lines may be required to support extended operation of the reactor coolant pumps since they supply cooling water to the reactor coolant pump enclosure air conditioning units, and

   b. The nitrogen supply line to the Pressurizer Relief Tank currently provides a redundant source of pneumatic motive power to the power-operated relief valves and will provide a similar function to their associated block valves.

   As discussed in our January 17, 1980 letter, these two systems will be modified to provide remote-manual containment isolation capability consistent with other essential systems. The revision of the "Containment Integrity" definition requires that the remote-manual containment isolation valves be operable, and allows them to be open during operating conditions which require "Containment Integrity."

   Based on our review of the Technical Specifications which might be affected by the implementation of the NRC TMI Short Term Lessons Learned Requirements, we have determined that the definition of "Containment Integrity" should have been revised as part of our thorough review of the San Onofre Unit 1 containment isolation design completed in

April, 1976 for the Sphere Enclosure Project and assessment of compliance with 10CFR50, Appendix J. As a result of this review, the containment isolation design was modified to improve the leak tightness capability of the containment in the event of an accident requiring containment isolation by installation of new automatic and remote-manual valves. These new valves, as well as existing remote-manual valves, were designated as containment isolation valves. The remote-manual valves provide the ability to isolate essential systems, if necessary, following an accident to improve the leak tightness of containment. However, the definition of "Containment Integrity" was not revised to reflect the improved containment isolation design utilizing both automatic and remote-manual valves as containment isolation valves.

2. The auxiliary feedwater flow test requirement and condensate tank level calibration and test requirements added to Table 4.1.1 specify the minimum frequency and type of surveillance to be applied to the instrumentation. These calibration and test requirements are consistent with existing surveillance requirements for instrumentation installed in other systems.

3. The revision to Containment Design features (Technical Specification 5.2) accurately describes the containment isolation design.

Based on the above, it is concluded that (1) the proposed change does not involve an unreviewed safety question as defined in 10CFR50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.