LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1	0 5 0 0 0 2 0 6	1 of 0 5

Title (4)

INADEQUATE APPLICABILITY FOR TECHNICAL SPECIFICATIONS RELATING TO CORE POWER DISTRIBUTION MONITORING DUE TO ENGINEERING DEFICIENCIES

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
									NONE	0 5 0 0 0
0 2	2 2	8 9	8 9	0 1 0	0 0					0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	___ 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	X Other (Specify in Abstract below and in text)
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	VOLUNTARY

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER (AREA CODE)
R. W. Krieger, Station Manager	7 1 4 3 6 8 - 6 2 5 5

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) XX NO

Expected Submission Date (15): Month ___ Day ___ Year ___

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On February 22, 1989, with Unit 1 in Mode 5, following an SCE review of both the Cycle 10 Reload Safety Evaluation Report and related correspondence from Westinghouse, the potential operation of Unit 1 in an unanalyzed condition was identified. The applicability of Technical Specifications (TSs) 3.10 and 3.11 (Mode 1 > 90% power), which provide requirements for monitoring of core power distribution, had allowed operation of the core at power levels below 90% without monitoring for power asymmetry. Westinghouse could not document analyses which assured acceptable results for a design basis event with initial conditions resulting from an unlikely occurrence which: 1) adversely affects core power distribution and 2) remains undetected.

SCE believes that this condition was the result of previously identified general engineering deficiencies, including excessive reliance on contractor work.

The applicability for TSs 3.10 and 3.11 has been changed from "Mode 1 above 90% Rated Thermal Power" to "Mode 1". This ensures that assumptions made in the safety analyses will remain valid by requiring the core power distribution to be monitored any time the unit is at power. As discussed in previous submittals to the NRC, SCE has taken aggressive corrective actions associated with the general engineering deficiencies mentioned above. These actions have included: 1) implementation of a design basis documentation program; 2) consolidation of engineering organizations; and 3) implementation of training programs.

There has been no indication that Unit 1 has ever operated with a core power distribution that was not considered in the safety analyses; therefore, this issue represents no safety significance.

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Plant: San Onofre Nuclear Generating Station
Unit: One
Reactor Vendor: Westinghouse
Event Date: 2/22/89

A. CONDITIONS AT TIME OF THE EVENT:

Mode 5 (Cold Shutdown) during the Cycle 10 refueling outage.

B. BACKGROUND INFORMATION:

Technical Specification (TS) 3.10, "Incore Instrumentation", and TS 3.11, "Continuous Power Distribution Monitoring", together ensure that the incore instrumentation will be utilized to take periodic measurements such that core power distribution, including axial offset, remains within acceptable values. Other information, including a correlation between the excore and incore instrumentation, are also required to be taken. The purpose of these TSs is to ensure that acceptable safety margins are maintained (by placing a limitation on power level) in the event that these measurements cannot be made within the specified surveillance interval.

A review of the history of TSs 3.10 and 3.11 indicates that they were incorporated into the TSs as a result of concerns about the power peaking factors associated with the potential phenomenon of fuel densification. This phenomenon causes fuel pellet shrinkage which, in combination with other factors, could result in gaps between pellets in the fuel column, causing power peaking factors to increase in the vicinity of the fuel pellets adjacent to the gaps.

In preparation for the Cycle 4 reload in 1973, Westinghouse prepared a safety analysis which considered the effects of fuel densification. This analysis was subsequently submitted to the AEC for review. As a result, the AEC required that SONGS 1 steady state power "not exceed 90% of 1347 MWt since the increase in peaking factors associated with the gaps caused by fuel densification would be less than ten percent" (Reference letter from D. J. Skovholt [AEC] to J. B. Moore [SCE], 7/20/73). This power operation limitation was later reflected in the action statements for TSs 3.10 and 3.11, which originally required that if power distribution measurements could not be made by the incore instrumentation, power operation was limited to 90% of full power. In 1988, this requirement was additionally reflected in the applicability of TSs 3.10 and 3.11, which was changed to "Mode 1 above 90% Rated Thermal Power". However, from the time of initial implementation of these TSs to the Cycle 10 refueling outage (prior to Cycle 10 operation), no detailed reviews had been made of the range of the TS applicability.

The incore monitoring system was installed (prior to initial plant operation) in order to permit raising the power rating of the plant from 395 MWe to 450 MWe. The system was "not intended to serve as continuous on-line instrumentation but to provide supplementary core monitoring to be

used from time to time to confirm reactor performance." (Reference WCAP 3269-46, "Incore Instrumentation...", March 1965.) Therefore, it can be concluded that the core was initially approved for operation based on analysis of expected operating conditions, imposition of various administrative controls to control power distribution (such as control rod insertion limits), and periodic incore monitoring (when operating at greater than 395 MWe).

This philosophy apparently was carried forward beyond Cycle 4 operation, when the concern for fuel densification caused the initial implementation of TSs 3.10 and 3.11.

As a result of the additional steam generator tubes plugged during the Cycle 10 refueling outage (prior to Cycle 10 operation), it was necessary to change the equation for axial offset in Technical Specification 3.11. This change was reviewed as part of the Westinghouse Cycle 10 Reload Safety Evaluation, which also included a re-evaluation of the bases for TS 3.10 and 3.11. This evaluation indicated that the applicability of Technical Specification 3.10 and 3.11 for only Mode 1 above 90% power could no longer be supported. This information was discussed in a submittal to the NRC dated March 17, 1989, "Technical Issues Impacting San Onofre Unit 1 Restart." This voluntary LER is being provided consistent with the commitment made in that March 17 submittal.

C. DESCRIPTION OF THE EVENT:

1. Event:

On February 22, 1989, with Unit 1 in Mode 5, following an SCE review of both the Cycle 10 Reload Safety Evaluation Report and related correspondence from Westinghouse, the potential operation of Unit 1 in an unanalyzed condition was identified. The applicability of TSs 3.10 and 3.11 (Mode 1 > 90% power) had allowed operation of the core at power levels below 90% without monitoring for power asymmetry. Westinghouse could not document analyses which assured acceptable results for a design basis event with initial conditions resulting from an unlikely occurrence which: 1) adversely affects core power distribution and 2) remains undetected.

2. Inoperable Structures, Systems or Components that Contributed to the Event:

None

3. Sequence of Events:

Not Applicable

4. Method of Discovery:

Refer to Section B.

5. Personnel Actions and Analysis of Actions:

Not Applicable

6. Safety System Responses:

Not Applicable

D. CAUSE OF THE EVENT:

Although the specific root cause for this event cannot be conclusively determined due to the intervening 17 years, SCE believes that this condition was the result of previously identified general engineering deficiencies, including excessive reliance on contractor work.

E. CORRECTIVE ACTIONS:

1. Corrective Actions Taken:

a. The applicability for TSs 3.10 and 3.11 has been changed from "Mode 1 above 90% Rated Thermal Power" to "Mode 1". This ensures that assumptions made in the safety analyses will remain valid by requiring the core power distribution to be monitored any time the unit is at power.

b. As discussed in previous submittals to the NRC, SCE has taken aggressive corrective actions associated with the general engineering deficiencies mentioned above. These actions have included: 1) implementation of a design basis documentation program; 2) consolidation of engineering organizations; and 3) implementation of training programs.

F. SAFETY SIGNIFICANCE OF THE EVENT:

The operability requirements for the incore instrumentation and the limits of power distribution are necessary to assure that the core physics are maintained in a manner consistent with the assumptions in the applicable safety analyses. Although sufficient documentation could not be identified to support the "Mode 1 above 90% power" applicability of TSs 3.10 and 3.11, the AEC had stated upon the initial issuance of TS 3.10 that "the proposed surveillance of the core power distribution will provide the necessary information for verifying that the maximum power density is being maintained below the allowable limit or that acceptable safety margins will be provided in the event that...measurements cannot be made..." (Reference letter from Donald J. Skovholt [AEC] to Jack B. Moore [SCE], 11/13/73.) Assuming this statement was based on valid information (which could not be verified), operation within the safety analyses was assured.

In the past, it has been SCE's practice to perform the measurements required by TSs 3.10 and 3.11 at all power levels. It has never been necessary to reduce Unit 1 reactor power below 90% because of exceeding the axial offset limits. On one occasion, however, during Cycle 8 operation, power was reduced to less than 90% for a short time due to an inability to perform the incore-excore calibration required by TS 3.10 in a timely manner. The plant operated stably during this period. When the incore-excore instrumentation was subsequently calibrated and the axial offset measurements performed, there was no indication that adverse changes had occurred in core power distribution.

Therefore, there has been no indication that Unit 1 has ever operated with a core power distribution that was not considered in the safety analyses. Thus, this issue represents no safety significance.

G. ADDITIONAL INFORMATION:

1. Component Failure Information:

Not Applicable

2. Previous LERs for Similar Events:

Several LERs have been submitted which involved similar design and design control deficiencies. The most recent of these LERs include:

LER 90-010 (Docket No. 50-206) reported an error in the spent fuel pool decay heat calculation.

LER 89-012, Revision 1 (docket No. 50-361) reported instrument mounting configuration discrepancies.

Corrective actions implemented from these events are similar to those reported in this LER. However, since these corrective actions were implemented subsequent to the event being reported in this LER, they could not have prevented this event from occurring.