UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	Docket No. 50-206
	)	
SOUTHERN CALIFORNIA EDISON COMPANY	)	License No. DPR-13
	)	
SAN DIEGO GAS AND ELECTRIC COMPANY	)	
	)	
San Onofre Nuclear Generating Station, Unit No. 1	)	

ORDER FOR MODIFICATION OF LICENSE

I.

Southern California Edison Company and San Diego Gas and Electric Company (the licensees) are the holders of Provisional Operating License No. DPR-13 which authorizes the licensees to operate San Onofre Nuclear Generating Station, Unit No. 1, at power levels up to 1347 megawatts thermal (rated power). The facility is a pressurized water reactor located on the licensees' site in San Diego County, California. The license is subject to all applicable provisions of the rules, regulations, and orders of the Nuclear Regulatory Commission.

II.

Following the accident at the Three Mile Island Unit 2 facility on March 28, 1979, a number of investigations were undertaken to assess the adequacy of design features, operating procedures, and personnel of nuclear power plants to provide assurance of no undue risk regarding severe reactor accidents. The report "NRC Action Plan Developed as a Result of the TMI-2 Accident" (NUREG-0660, May 1980) described a comprehensive and integrated plan involving many actions that served to increase safety. The Commission-approved items for implementation were identified in a second report "Clarification of TMI Action Plan Requirements" (NUREG-0737, November 1980).

Among the items approved was item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling" which required licensees to describe additional means proposed to provide an unambiguous, easy-to-interpret, indication of inadequate core cooling (ICC).

On November 4, 1982, the Commission refined its requirements in this area and determined that an instrumentation system for detection of inadequate core cooling (ICC) consisting of upgraded subcooling margin monitors, core-exit thermocouples, and a reactor coolant inventory tracking system is required for the operation of pressurized water reactor facilities. These new requirements were the subject of Generic Letter No. 82-28, "Inadequate Core Cooling Instrumentation System" (December 10, 1982).

On the basis of analysis of information provided by licensees, meetings with industry groups and independent studies by the NRC Staff, the Commission found that during a small LOCA, there is a period of time before the core has boiled dry (indicated by core exit thermocouples) when the operators have insufficient information to clearly indicate a void formation in the reactor vessel head or to track the inventory of coolant in the vessel and primary system. The Subcooling Margin Monitor gives early indication of a problem but does not indicate whether the condition is getting worse or better.

The addition of a reactor coolant inventory system improves the reliability of plant operators in diagnosing the approach of ICC and in assessing the adequacy of responses taken to restore core cooling. The benefit is preventive in nature in that the instrumentation assists the operator in avoidance of ICC when voids in the reactor coolant system and saturation conditions result from overcooling events, steam generator tube ruptures, and small break loss of coolant events. The addition of a reactor coolant inventory system, coupled with upgraded in-core thermocouple instruments and a subcooling margin monitor, provides an ICC instrumentation package which could significantly reduce the likelihood of human misdiagnosis and errors for events such as steam generator tube ruptures, loss of instrument bus or control system upsets, pump seal failures, or overcooling events originating from disturbances in the secondary coolant side of the plant. For events of lower likelihood, involving coincidental multiple faults or more rapidly developing small break LOCA conditions, the ICC also reduces the probability of human misdiagnosis and subsequent errors leading to ICC.

III.

Southern California Edison (SCE) responded to Generic Letter 82-28 on June 20, 1986,* in which SCE concluded that no additional ICC instrumentation was required, but that the core-exit thermocouple system would be upgraded to

*The response was delayed due to an extended outage to upgrade the seismic design of the facility.

meet requirements of NUREG-0737. Thus, SCE did not propose to install a reactor vessel water level instrument, contrary to Generic Letter 82-28. The NRC staff's review of the SCE submittal was issued on May 9, 1988 which concluded that relief from this post-TMI item was not acceptable and requested SCE to respond with a commitment to install a reactor vessel water level instrumentation system (RVLIS) and respond to a request for additional information concerning other aspects of II.F.2 - the core-exit thermocouples and the subcooling margin monitor.

By letter of August 8, 1988, SCE responded to the request for additional information. In its response, SCE reiterated its belief that there are insufficient safety or technical benefits to merit the large expediture, but provided additional information regarding a proposed single non-safety grade thermocouple which could be installed in the reactor upper internals.

SCE's relief from the requirement to install a RVLIS is based upon unique plant design, plant response to transients, and offsite dose considerations.

In comparison to larger and more modern plants, SONGS-1 has a smaller core (1347 Mwt) resulting in a smaller source term. The margin between operating pressure (2085 psi) and the PORV setpoint (2190 psi) is the same as other plants (100 psi). The margin between normal operating temperature and the saturation temperature for the RCS is about 10°F greater. Due to its smaller RCS volume (6940 ft^3) compared with large plants (10,600 ft^3) and high capacity injection pumps, the licensee stated that SONGS-1 is less susceptible to small break LOCAs.

SCE evaluated the Westinghouse differential pressure (dp) RVLIS and the Combustion Engineering (CE) Heated Junction Thermocouple (HJTC) systems for their applicability to SONGS-1. The Westinghouse RVLIS generic design cannot be installed in SONGS-1 due to the lack of a bottom vessel penetration. SCE stated that the pressure vessel head penetrations and reactor internals would require extensive modification and corresponding hydraulic and stress analyses in order to accommodate HJTC probes, resulting in very high cost. Their value-impact assessment was done using the estimated costs associated with installing the HJTC system (this cost is two to three times higher than the average cost). SCE further stated that it has considered a modified Westinghouse dp system which would measure only from the hot leg to the vessel head, but did no serious cost estimates for this alternative. The alternative head-to-hot leg design was deemed less desirable because of its limited measurement range, which is essentially the same as the HJTC system, and its inoperability when primary coolant pumps are running. No mention was made of the possibility of using pump power monitoring to detect voids when the pumps are running. This alternative would meet II.F.2 requirements.

SONGS-1 is the smallest of five Westinghouse plants that are in the range of 0.4 to 0.5 times the size of the modern four-loop Westinghouse plant. Only one, Yankee Rowe, is smaller (600 Mwt) and it has been granted relief from installing a RVLIS. Yankee Rowe, which is located in a remote area, is operating at 400 psi margin between operating pressure and the PORV setpoint and 25°F subcooling margin with upgraded SMM and CET systems. Haddam Neck and Ginna are the most similar to SONGS-1 due to the fact that they do not have reactor vessel bottom penetrations. Ginna has installed a modified Westinghouse type RVLIS from head-to-hot leg and Haddam Neck has installed the CE HJTC system.

A considerable portion of the SCE submittal is devoted to evaluation of events potentially leading to inadequate core cooling and a probabilistic risk assessment to evaluate the risk of core damage due to ICC with the existing SONGS-1 design. Although the study has merit, it involves essentially the same considerations used by the licensees and the NRC several years ago in determining if additional ICC requirements were needed. After due consideration of many analyses, the NRC determined that new requirements were needed and imposed those defined in NUREG-0737. The evaluation of the SCE exemption request is therefore based on the differences in performance between SONGS-1 and larger plants that make ICC less likely to occur or the consequences less severe.

The principal difference noted by the scenarios presented in the submittal is that the smaller RCS volume of SONGS-1 leads to more rapid depressurization and early initiation of safety injection flow in the event of a small break. Safety injection flow can match the break flow for breaks smaller than 2.5 inches and core uncovery is not predicted. For larger breaks in the 2.5 to 5.2 inch range, less uncovery and lower peak clad temperatures are predicted than for larger plants. The report suggests that a 3.0 inch line break at SONGS-1 is comparable to a 4.0 inch line break at more typical larger reactors.

A probabilistic risk assessment focused on the risk associated with inventory-threatening scenarios leading to inadequate core cooling. In a small break LOCA event, the safety injection system is designed to provide sufficient injection flow to prevent core damage. When this injection flow has depleted the Refueling Water Storage Tank, the system must be realigned so that the normal charging system in conjunction with the containment recirculation pumps provide recirculation and injection to the reactor coolant system. The risk assessment utilizes event and fault trees to describe the sequence of events and failures which must occur for an inadequate core cooling condition to result. The fault trees include both equipment and human operator performance estimates for the existing plant configuration and instrumentation. The data input for the assessment were taken mostly from other studies. The validity of this data has not been confirmed, but the overall result appears reasonable in relation to the information given for other plants.

The risk assessment was then adjusted for an assumed incremental improvement which might result from the inclusion of RVLIS. It was assumed, however, that the RVLIS did not provide any new information of value, but was only redundant to the existing information so that the scenarios were virtually unaffected by the addition of RVLIS information. No procedural adjustments were made based on the availability of RVLIS. The only indication of approach to ICC in the existing system is elevated hot leg temperatures or elevated core-exit temperatures which respond only late in the event when the core is partially uncovered. The evaluation gives no credit for early detection of approaching ICC with vessel water level indication and the corresponding potential for corrective operator action. This is contrary to the conclusion reached by the staff based on other analyses which led to the establishment of the RVLIS requirements. The biased input assumption that RVLIS is of no value guarantees the PRA result obtained. For each of the assessments, it is stated that the uncertainty of the analysis was not calculated, but is assumed to be large. In fact, the value of RVLIS in significantly increasing the likelihood of correct operation response is not adequately assessed.

The offsite consequence analysis submitted by SCE does not present a clear case for any less diligence in detecting and preventing ICC. Recent increase in population density in the vicinity of the plant appears to place SONGS-1 in a similar risk category as many other plants, all of which have installed ICC instrumentation as mandated. SONGS-1 has a small advantage due to its lower operating power and correspondingly smaller source term.

The licensees have not presented adequate technical justification for excluding a reactor vessel level measurement system from their instrumentation to detect inadequate core cooling. Other plants of similar size and design have installed an acceptable level measurement system (either a generically approved one or a modified one). For the reasons discussed, I have determined that the public health, safety, and interest require implementation of item II.F.2 of the TMI Action Plan and that the license should be modified, as described below.

IV.

Accordingly, pursuant to Sections 103, 161b and 161i of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.204 and 10 CFR Part 50, IT IS HEREBY ORDERED that Provisional Operating License No. DPR-13 is hereby modified as follows:

> Licensees shall implement all the requirements of item II.F.2, "Inadequate Core Cooling Instrumentation System" (Generic Letter 82-28) as soon as practicable but not later than startup for fuel cycle XI (approximately January 1991). Specific plans for implementation shall be submitted to NRC for approval by no later than December 1, 1989.

The licensees or any person who has an interest adversely affected by this order may request a hearing within 30 days of the date of publication of this order in the FEDERAL REGISTER. A request for hearing must be addressed to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with copies to the Assistant General Counsel for Enforcement at the same address. If a person other than the licensees requests a hearing, that person shall set forth with particularity the manner in which the petitioner's interest is adversely affected by this order and should address the criteria set forth in 10 CFR 2.714(d).

If a hearing is requested, the Commission will issue an order designating the time and place of the hearing. If a hearing is held, the issue to be considered shall be whether this order should be sustained. Upon the failure to answer or request a hearing within the specified time, this order shall be final without further proceedings.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
this 10th day of May, 1989