DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS
PROPOSED CHANGE NO. 96 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE DPR-13

This is a request to revise Appendix A Technical Specificatons 1.0 Definitions, 3.1 Reactor Coolant System, 3.5 Instrumentation and Control, 3.6 Containment, 3.8 Fuel Loading and Refueling, and 4.1 Operational Safety Items. Additionally, this a request to revise Proposed Change No. 89 and Proposed Change No. 93.

Reason for Proposed Change

Subsequent to their review of decay heat removal capability at operating PWR facilities, the NRC, on June 11, 1980, requested all operating PWRs to amend the Technical Specifications with respect to decay heat removal capability; specifically, the Division of Licensing requested proposed changes that provide for redundancy in decay heat removal capability in all modes of operation. To comply with this NRC request, three existing Technical Specifications require extensive revision. Additionally, modes of operation have been defined to conform more closely to the Standard Technical Specifications; as a result, two existing Technical Specifications and two existing Proposed Changes require revision to be compatible with the definitions of modes of operation.

Existing Specifications

The existing specifications are as constituted in Sections 1.0 Definitions, 3.1 Reactor Coolant System, 3.5 Instrumentation and Control, 3.6 Containment, 3.8 Fuel Loading and Refueling, and 4.1 Operational Safety Items.

Proposed Specifications

The existing specifications and existing proposed changes would be revised as indicated in the Enclosure to this Proposed Change. The added or revised portions are identified by a bar in the margin.

Safety Analysis

The Technical Specification Changes discussed in the enclosure are provided to ensure that proper means are available to provide redundant methods of decay heat removal in all modes of operation. For all modes of operation except for sufficient water in the refueling pool during refueling, the single failure criterion is imposed to ensure redundant decay heat removal capability. When there is sufficient water in the refueling pool during refueling, a large heat sink is available for core cooling; thus, in the event of a failure of the operating residual heat removal loop, adequate time is available to initiate alternate means to cool the core. The Bases of the Proposed Change provide greater detail to support and supplement this Safety Analysis.

Accordingly, 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.

ENCLOSURE

The following definition will be added to Section 1.0:

Operational Mode - Mode

An Operational Mode (i.e., Mode) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$>$ 5%	$\geq 350^{\circ}F$
2. STARTUP	≥ 0.99	$\leq$ 5%	$\geq 350^{\circ}F$
3. HOT STANDBY	< 0.99	0	$\geq 350^{\circ}F$
4. HOT SHUTDOWN	≤ 0.95	0	$350^{\circ}F > T_{avg} > 200^{\circ}F$
5. COLD SHUTDOWN	≤ 0.95	0	$\leq 200^{\circ}F$
6. REFUELING**	≤ 0.95	0	$\leq 140^{\circ}F$

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

3.1.2 OPERATIONAL COMPONENTS

Applicability: Applies to the operating status of the reactor coolant system equipment and related equipment.

Objective: To identify those conditions of the reactor coolant system necessary to ensure safe reactor operation.

Specification: A. At least one pressurizer safety valve shall be operable or open when the reactor head is on the vessel, except for hydrostatic tests.

B. The concentration of boron in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one residual heat removal pump is circulating reactor coolant.

C. The reactor shall not be made critical or maintained critical unless both pressurizer safety valves are operable.

D. During Startup and Power Operation, all three reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation. With less than the above required coolant loops in operation, be in Hot Standby within 1 hour, except as modified by Specification E.

E. The limitations of Specification D may be suspended during Startup and Power Operation as follows:

(1) Operation may be conducted with 0, 1, 2, or 3 reactor coolant pumps operating during low power physics testing at less than 5% of full power.

(2) Whenever reactor power is less than 10% of full power, operation with one or two reactor coolant pumps operating shall be limited to less than 24 consecutive hours.

F. During Hot Standby the following specifications shall apply:

(1) At least two of the reactor coolant loops listed below shall be Operable:

a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

(2) At least one of the above coolant loops shall be in operation.*

(3) With less than the above required reactor coolant loops Operable, restore the required loops to Operable status within 72 hours or be in Hot Shutdown within the next 12 hours.

(4) With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required coolant loop to operation.

G. During Hot Shutdown and Cold Shutdown, the following specifications shall apply:

(1) At least two of the coolant loops listed below shall be Operable:

a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

d. Residual heat removal (RHR) pump G-14A and one RHR heat exchanger.**

e. Residual heat removal (RHR) pump G-14B and one RHR heat exchanger.**

(2) At least one of the above coolant loops shall be in operation.***

*All reactor coolant pumps may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

**The normal or emergency power source may be inoperable in the Cold Shutdown condition.

***All reactor coolant pumps and residual heat removal pumps may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

(3) With less than the above required loops Operable, immediately initiate corrective action to return the required loops to Operable status as soon as possible; if in Hot Shutdown, be in Cold Shutdown within 20 hours.

(4) During Cold Shutdown, when the reactor coolant loops are not Operable, one RHR pump and one RHR heat exchanger may be removed from Operable status for maintenance or replacement of components provided that the core outlet temperature is maintained less than 190°F, but for a period of time not to exceed 72 hours.

(5) With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required coolant loop to operation.

Basis: One pressurizer safety valve is sufficient to prevent over-pressurizing when the reactor is subcritical, since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., residual heat, pump energy and pressurizer heaters.

Prior to reducing boron concentration by dilution with make up water either a reactor coolant pump or a residual heat removal pump is specified to be in operation in order to provide effective mixing. During boron injection, the operation of a pump, although desirable, is not essential. The boron is injected into an inlet leg of the reactor coolant loop. Thermal circulation which exists whenever there is residual heat in the core, will cause the boron to flow to the core.

Lack of further mixing cannot result in areas of reduced boron concentration within the core. Prior to criticality the two pressurizer safety relief valves are specified in service in order to conform to the system relief capabilities.(1)

Heat transfer analyses show that reactor heat equivalent to 8% of full power can be removed with natural circulation only; hence, for up to 24 hours the specified upper limit of 10% of full power with 1 or 2 reactor coolant pumps operating provides a substantial safety factor.

With three reactor coolant pumps in operation, the DNB ratio would not drop below 1.30 after a loss of flow with a reactor trip.(2)(3)

In Hot Standby, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be Operable.

In Hot Shutdown and Cold Shutdown, a single reactor coolant loop or the residual heat removal system provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be Operable. Thus, if the reactor coolant loops are not Operable, this specification requires both residual heat removal pumps and both residual heat exchangers to be Operable. However, since all RHR components are inside containment, maintenance on the RHR system can be accomplished only during shutdown; therefore, exception to requiring the Operability of two RHR pumps and two RHR heat exchangers is necessary. This requires that either (1) decay heat is sufficiently low that core outlet temperature will not exceed 190°F in the event that the operating RHR pump fails, or (2) a temporary or replacement pump is available for installation with sufficient allowance so that core outlet temperature does not exceed 190°F. The temperature of 190°F is selected so that Cold Shutdown conditions are not exceeded and to provide sufficient margin to saturation temperature. The specified maintenance time is a maximum, and maintenance work will proceed with diligence to return the equipment to Operable condition as promptly as possible. The specified maintenance time is a reasonable time to expect the operating RHR pump to continue operating while maintenance is performed.

The operation of one reactor coolant pump or one residual heat removal pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the reactor coolant system. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

References: (1) Final Engineering Report and Safety Analysis, Sections 9 and 10.

(2) Final Engineering Report and Safety Analysis, Paragraph 10.2.

(3) Supplement No.1 to Final Engineering Report and Safety Analysis, Section 3, Question 9.

3.8 Fuel Loading and Refueling

Applicability: Applies to fuel loading and refueling operations.

Objective: To prevent incidents during fuel handling operations that could affect public health and safety.

Specification: A. During refueling operations:

(1) Radiation levels in the containment and spent fuel building shall be monitored.

(2) Core subcritical neutron flux shall be continuously monitored during the entire refueling period by not less that two neutron monitors, each with continuous visual indication and one with continuous audible indication.

(3) For all water levels in the refueling pool, the following specifications shall apply:

a. At least one of the following methods of decay heat removal shall be in operation:

1. One residual heat removal pump and one residual heat removal heat exchanger.

2. One refueling water pump, taking suction from the refueling pool through the recirculation heat exchanger, and discharging via the safety injection system piping to one reactor coolant loop cold leg.

b. With less than one method of decay heat removal in operation, except as provided in c. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system.

c. The decay heat removal capability may be removed from operation for up to one hour per eight hour period.

(4) During the time that the reactor vessel head is detensioned with bolts in place, and whenever the water level in the refueling pool is less than elevation 40 feet 3 inches [see (7) below for reference elevation], the following specifications shall apply:

a. Two of the following methods of decay heat removal shall be Operable:

1. Residual heat removal (RHR) pump G-14A and one RHR heat exchanger.

2. Residual heat removal (RHR) pump G-14B and one RHR heat exchanger.

3. One refueling water pump, lined up to take suction from the refueling pool through the recirculation heat exchanger, and lined up to discharge via the safety injection system piping to one reactor coolant loop.

b. With less than the required methods of decay heat removal Operable, immediately initiate corrective action to return the required methods of decay heat removal to Operable status as soon as possible.

(5) During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 2900 ppm shall be maintained in the primary coolant system.

(6) The reactor shall be subcritical for at least 148 hours prior to movement of irradiated fuel in the reactor pressure vessel.

(7) Water borated as specified in Item A.(5) above shall be maintained at an elevation not less than 40 feet 3 inches in the refueling pool during movement of fuel assemblies and RCC's. Reference elevation is sea level, mean lower low water.

(8) If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

B. With fuel assemblies in the spent fuel storage pool:

(1) Loads in excess of 1,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

(2) Water borated as specified in Item A(5) above shall be maintained at an elevation not less than 40 feet 3 inches in the spent fuel storage pool. Reference elevation is sea level, mean lower low water.

(3) With the requirement of B(2) above not satisfied suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limits within four hours.

Basis:

During refueling the reactor refueling cavity is filled with approximately 240,000 gallons of borated water of 2,900 ppm boron concentration. This boron concentration is sufficient to maintain the reactor subcritical approximately by 10% $\Delta k/k$ with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.(1) Operation of one method of decay heat removal is provided to assure continuous mixing flow of refueling water through the reactor vessel during the refueling period.(2) Borated water injection capability is provided as per Specification 3.2 Part A in the unlikely event there is any need during the refueling period.

The requirement that at least one method of decay heat removal be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during Refueling, and (2) sufficient cooling circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two methods of decay heat removal Operable when the refueling pool water level is less than elevation 40 feet 3 inches ensures that a single failure of an operating component will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling; thus, in the event of a failure of an operating component, adequate time is available to initiate alternate means to cool the core.

In addition to the above safeguards interlocks are utilized during refueling to insure safe handling.[3] These include:

(1) An interlock on the lifting hoist to prevent lifting of more than one fuel assembly at any one time.

(2) The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The restriction on movement of loads in excess of 1,500 pounds (i.e., the nominal weight of a fuel assembly, RCC, and associated handling tool) over fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with activity release assumed in the accident analysis.

Requiring a minimum water elevation of 40 feet 3 inches in the refueling pool, and similarly in the spent fuel storage pool, ensures that 1) at least 23 feet of water would be available to remove 99% of the iodine gap activity assumed to be released in the event of a dropped and damaged fuel assembly, and 2) there will be at least twelve feet of water above the top of the fuel rods of a withdrawn fuel assembly so as to limit dose rates at the top of the water in accordance with Section 4.2.6 of the facility FSA. Reference elevation is sea level, mean lower low water.

Finally, detailed written procedures are provided, and are carried out under close supervision by licensed personnel.

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel assures that sufficient time has elapsed to allow the radioactive decay of short-lived fission products.

References: (1) Final Safety Analysis, Paragraph 2.4.

(2) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 5, Question 8 and 9.

(3) Final Safety Analysis, Paragraph 2.9.

TABLE 4.1.2 (Continued)

			Check	Frequency
11.	MOV/LCV-1100 C Transfer Switch	a.	Verify that the fuse block for either breaker 8-1198 to MCC 1 or breaker 42-12A76 to MCC 2A is removed.	Same as Item 10 above
12.	Emergency Siren Transfer Switch	a.	Verify that the fuse block for either breaker 8-1145 to MCC 1 or breaker 8-1293A to MCC 2 is removed.	Same as Item 10 above
13.	Communication Power Panel Transfer Switch	a.	Verify that the fuse block for either breaker 8-1195 to MCC 1 or breaker 8-1293B to MCC 2 is removed.	Same as Item 10 above
14.a.	Spent Fuel Pool Water Level	a.	Verify water level per Technical Specification 3.8.	a. Once every seven days when spent fuel is being stored in the pool.
b.	Refueling Pool Water Level	b.	Verify water level per Technical Specification 3.8.	b. Within two hours prior to start of and at least once per 24 hours thereafter during movement of fuel assemblies or RCC's.
15.	Reactor Coolant Loops	a.	Verify that all three loops are in operation and circulating reactor coolant.	a. Every 12 hours during Startup and Power Operation.
		b.	Verify Operability per Technical Specification 3.1.2.F.(1)	b. At least the 2 required reactor coolant pumps, if not in operation, shall be determined Operable once per 7 days by verifying correct breaker alignments and indicated power availability during Hot Standby.

		c.	Verify one coolant loop in operation and circulating reactor coolant per Technical Specification 3.1.2.F.(2)	c.	At least once per 12 hours in Hot Standby.
		d.	Determine that the required coolant loops are Operable per Technical Specification 3.1.2.G.(1)	d.	During Hot Shutdown and Cold Shutdown, the required reactor coolant pump(s), if not in operation, shall be determined Operable once per 7 days by verifying correct breaker alignments and indicated power availability. The required residual heat removal system components shall be determined Operable per Specification 4.7 and once per 7 days by verifying correct breaker alignments and indicated power availability
		e.	Verify at least one coolant loop in operation and circulating reactor coolant per Technical Specification 3.1.2.G.(2)	e.	At least once per 12 hours during Hot Shutdown and Cold Shutdown
16.	Decay Heat Removal capability	a.	Verify Operability of two methods of decay heat removal per Technical Specification 3.8.A.(4)a.	a.	The decay heat removal components shall be determined Operable per Specification 4.7

TABLE 3.5.1
INSTRUMENT OPERATING CONDITIONS

Functional Unit	COLUMN I Minimum Operational Channels	COLUMN II Minimum Redundancy* Required	COLUMN III Required Operating Action if Column I or Column II Cannot be Met
1. Nuclear Power-Critical	3	For 3-Channel Operation --1 For 4 Channel Operation --2	Maintain hot standby conditions.
-Subcritical	3	1	Maintain hot standby if at least one source and one intermediate channel are available; otherwise maintain 10% Δk/k shutdown margin.
2. Pressurizer Variable Low Pressure	2	1	Maintain load below 10% F. P.
3. Pressurizer Fixed High Pressure	2	1	Maintain hot standby conditions.
4. Pressurizer High Level	2	1	Maintain hot standby conditions.
5. Reactor Coolant Flow — 3-Loop Operation	3	1**/2***	Maintain load below 10% F. P.
6. Pressurizer Low Pressure (Safety Injection Function)	2	1	Maintain hot standby conditions.

* Redundancy is defined as N-M, where N is the number of channels in operation, and M is the number of channels in operation which, when tripped, will cause an automatic shutdown.

** For operation at ≤50% of full power
*** For operation at >50% of full power.

TABLE 3.5.1 (continued)

INSTRUMENT OPERATING CONDITIONS

Functional Unit	COLUMN I Minimum Operational Channels	COLUMN II Minimum Redundancy* Required	COLUMN III Required Operating Action if Column I or Column II Cannot be Met
7. Pressurizer Low Level (Safety Injection Function)	2	1	Maintain hot standby conditions.
8. Manual Trip	1		Maintain hot standby conditions.
9. Containment Sphere Pressure (Isolation Valve Signal)	1		Maintain cold shutdown.
10. Steam Feed-Water Flow Mismatch	3	1	Operator shall assume continuous surveillance and actuate manual scram if required.

* Redundancy is defined as N-M where N is the number of channels in operation, and M is the number of channels in operation which, when tripped, will cause an automatic shutdown.

3.5.2 Control Group Insertion Limits

Applicability: This standard applies to the insertion limits for the control banks during Startup and Power Operation.

Objective: To ensure (1) an acceptable core power distribution during power operation, (2) a limit on potential reactivity insertions for a hypothetical control rod ejection, and (3) core subcriticality after a reactor trip.

Specification:

A. The positions of the control rod groups shall be at or above the limits shown in Figure 3.5.2.1 except during low power physics tests.

B. The energy weighted average of the positions of control bank 2 shall be at least 90% withdrawn after the first 20% burnup of a core cycle. The average shall be computed at least twice every month and shall consist of all control bank 2 positions during the core cycle.

C. If it is determined that a rod has been dropped, retrieval shall be performed without increasing power level. An evaluation of the effect of the dropped rod shall be made to establish permissible power levels for continued operation. If retrieval is not successful within 3 hours, appropriate action, as determined from the evaluation, shall be taken. In no case shall operation longer than 3 hours be permitted if the dropped rod is worth more than 0.4% $\Delta k/k$.

D. No more than one inoperative rod shall be permitted during critical operation.

Basis: During Startup and Power Operation, the shutdown groups are fully withdrawn and control of the reactor is maintained by the control groups. The insertion limits are set in consideration of maximum specific power, shutdown capability, and the rod ejection accident. The considerations associated with each of these quantities are as follows:

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design maximum value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 13.97 kW/ft., 1.55 and 2.95, respectively. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

3.6 CONTAINMENT

Applicability: Applies to the operating status of the containment sphere.

Objective: To ensure containment integrity.

Specification: A. Leakage

The reactor coolant system temperature shall not be increased above 200 °F if the containment leakage exceeds the maximum acceptable values specified in Surveillance Standard 4.3.

B. Access to Containment

(1) Containment integrity shall not be violated unless the reactor coolant system is below 500 psig and a shutdown margin greater than 1% $\Delta k/k$ with all rods inserted is maintained for the most reactive temperature.

(2) Containment integrity shall not be violated when the reactor coolant system is open to the containment atmosphere unless a shutdown margin greater than 5% $\Delta k/k$ is maintained with all control rods inserted.

(3) Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact.

C. Internal Pressure

The reactor shall not be made critical, nor be allowed to remain critical, if the containment sphere internal pressure exceeds 0.4 psig, or the internal vacuum 2.0 psig.

Basis: The basis for the shutdown margins and 500 psig pressure are as follows:

$\Delta k/k$	Event	Basis for Adequacy
1% (below 500 psig)	Violation of Containment	Safety injection system disarmed; no credible automatic or operator action could cause return to criticality.

Δk/k	Event	Basis for Adequacy
5%	Open reactor coolant system	Provides adequate margin so that maintenance activities can be carried out with the reactor head removed.(1)

Regarding internal pressure limitations, the containment design pressure of 46.4 psig would not be exceeded if the sphere internal pressure before a major loss of coolant accident was no greater than 0.4 psig. The design criteria also allows an internal vacuum not in excess of 2.0 psig. Thus, the specified limiting conditions for internal pressure are consistent with the design basis.(2) Although such design values could be exceeded without damage to the structure, it is considered that the importance of the containment function warrants the specified values.

Opening of the ventilation system backup valves, POV 9A and POV 10A, is not considered a violation of containment integrity during startup conditions provided that their corresponding in-line valves POV 9 and POV 10 are closed.

References:

(1) Supplement No. 3 to Final Engineering Report and Safety Analysis, Question No. 2.

(2) Final Engineering Report and Safety Analysis, Paragraph 5.3.

DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS
PROPOSED CHANGE NO. 89 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE DPR-13

This is a request to revise Appendix A Technical Specifications 3.5, INSTRUMENTATION AND CONTROL and 4.1, OPERATIONAL SAFETY ITEMS.

Reason for Change

As a part of an NRC staff review of the LER's and Technical Specification requirements related to the Control Rod Position Indication Systems (RPI) at Westinghouse PWR's it was determined that a wide variation exists in the plant specific requirements in this area. In an effort to clarify the NRC requirements, a letter dated November 5, 1979 was provided to holders of an operating license which requested that the existing Technical Specifications be reviewed to ensure that the control rods are required to be maintained within margins of the Westinghouse safety analyses for control rod misalignment. Additionally an NRC staff decision was provided which deemed acceptable the use of direct LVDT voltage measurements in lieu of the position recorder indication to determine the operability of the Rod Position Indication System.

By letter dated January 4, 1980 we committed to implement the above described Technical Specification changes.

Existing Specification

The existing Technical Specifications 3.5 and 4.1 are as constituted in Appendix A to Provisional Operating License No. DPR-13.

Proposed Specification

Technical Specification 3.5.2, CONTROL GROUP INSERTION LIMITS, would be revised to read in part:

(1) "A. The position of all control rods shall be at or above the limits shown in Figure 3.5.2.1 except during low power physics tests."

(2) "D. Deleted."

The Basis for Technical Specification 3.5.2, CONTROL GROUP INSERTION LIMITS, would be revised by the deletion of the last paragraph which now reads, "Specification D provides that the control rod system will always be capable of shutting down the reactor."

Technical Specification 3.5 would be revised by the addition of Sections 3.5.3, CONTROL AND SHUTDOWN ROD MISALIGNMENT and 3.5.4, ROD POSITION INDICATION SYSTEM, to read as indicated in the Enclosure 1.

The balance of Technical Specification 3.5 would remain as constituted in Appendix A to Provisional Operating License No. DPR-13.

Technical Specification 4.1 would be revised by the addition of paragraphs E., F., and G., under Specification to read as follows:

"E. All control rods shall be determined to be above the rod insertion limits shown in Figure 3.5.2.1 by verifying that each analog detector indicates at least 21 steps above the rod insertion limits, to account for instrument inaccuracies, at least once per shift during Power Operation and at least once per shift during Startup conditions with K_{eff} equal to or greater than one.

F. The position of each rod shall be determined to be within the group demand limit and each rod position indicator shall be determined to be OPERABLE by verifying that the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) agree within 35 steps at least once per shift during Startup and Power Operation except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) at least once per 4 hours.

G. Each rod not fully inserted in the core shall be determined to be operable by movement of at least 10 steps in any one direction at least once per 31 days."

Table 4.1.2 Item 4 would be revised to read, "4. Deleted"

The balance of Technical Specification 4.1 would remain as constituted in Appendix A to Provisional Operating License No. DPR-13.

Safety Analysis

The Technical Specification Changes discussed above are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. Operability of the control rod position indicators is required to determine control rod alignment and insertion limits.

The conditional statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

Control rod positions and operability of the rod postion indicators are required to be verified on a nominal basis of once per shift with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisified.

As indicated in the Westinghouse report entitled, "Verification of Rod Misalignment Technical Specification, San Onofre Unit No. 1, Southern California Edison Company," dated March 1980 (Enclosure 2), it was determined that an individual rod can be misaligned up to 21 inches from the bank position without impacting safe operation as long as all control rods are maintained above the rod insertion limits. Since the San Onofre Unit 1 control rod step size is 3/8 inch, the 21 inches is equal to 56 steps.

The accuracy of the analog detection system is specified to be $\pm$ 17 steps in Section 5.4 of the San Onofre Unit 1 Final Safety Analysis. It has been determined that this accuracy applies to steady state operation and that during startup or shutdown maneuvers when thermal transients are enduced in the system, the accuracy of the instrument tends to drift. In order to determine the amount of drift which would be experienced by the analog detection system, a statistical evaluation of past rod position recorder data was performed. The results of this evaluation indicate that the analog detection system has an accuracy of $\pm$ 21 steps (with a 95% confidence limit) during periods when the system is undergoing thermal transients.

Since an individual rod can be misaligned by up to $\pm$ 56 steps without impacting safe operation (as indicated in Enclosure 2), and the accuracy under transient conditions for the analog detection system was determined to be $\pm$ 21 steps, safe operation is assured by requiring that the analog detection system be maintained with $\pm$ 35 steps of the bank position indicated by the digital step counter. In addition, in order to assure that all rods are above the insertion limit, the indicated position will be maintained 21 steps above the limit in order to account for instrument inaccuracies.

Based upon the analysis provided in this report, 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 hazards 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.

ENCLOSURE 1

3.5.3 CONTROL AND SHUTDOWN ROD MISALIGNMENT

Applicability: Applies to the number of steps an individual control or shutdown rod may be misaligned from its group position during Startup and Power Operation.

Objective: To ensure that the effects of rod misalignment from the group position do not exceed the core design margins.

Specifications:

A. During Startup and Power Operation, all rods shall be OPERABLE and maintained within ± 35 steps (indicated by the Analog Detection System) of their step counter indicated bank position (indicated by the Digital Detection System), except during low power physics tests.

B. With Specification A, above, not met, the following specifications are applicable.

1. With one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN BASIS of Specification 3.5.2 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

2. With more than one rod inoperable or misaligned from the step counter indicated position by more than ± 35 steps (indicated by the Analog Detection System), be in HOT STANDBY within 6 hours.

3. With one rod inoperable due to causes other than addressed by Specification B.1, above, or misaligned from its step counter indicated height by more than ± 35 steps (indicated by the Analog Detection System), POWER OPERATION may continue provided that within one hour either:

   a. The rod is restored to OPERABLE status within the above alignment requirements, or

   b. The rod is declared inoperable and the SHUTDOWN MARGIN BASIS of Specification 3.5.2 is satisfied. POWER OPERATION may then continue provided that:

1) A reevaluation of each accident analysis of Table 3.5.3-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

2) The SHUTDOWN MARGIN BASIS of Specification 3.5.2 is determined at least once per 12 hours.

3) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and F^N_H are verified to be within their limits within 72 hours.

TABLE 3.5.3-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characterisitics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

4) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

5) The remainder of the rods in the group with the inoperable rod are aligned to within ± 35 steps of the inoperable rod within one hour while maintaining the rod insertion limits of Figure 3.5.2.1.

__Basis:__ The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential affects of rod misalignment on associated accident analyses.

The misalignment allowance of Specification B, assures core performance within allowed design margins including allowance for the inaccuracy of the position signals.

3.5.4 ROD POSITION INDICATING SYSTEM

Applicability: Applies to the operating status of the Rod Position Indicating System.

Objective: To ensure the ability to accurately detect the position of control and shutdown rods.

Specification:

A. During Startup and Power Operation the Analog Detection System and the Digital Detection System shall be OPERABLE and capable of determining the control rod positions within ± 21 steps.

B. The Analog Detection System remains OPERABLE if the specified rod position indications can be obtained from direct LVDT voltage measurements.

C. With specifications A and B, above, not met, the following specifications are applicable.

1. With a maximum of one rod position indicator (Analog Detection System) per bank inoperable either:

a. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors within 8 hours, and at least once per 8 hours thereafter and immediately after any motion of the non-indicating rod which exceeds 56 steps in one direction since the last determination of the rod's position, or

b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

2. With a maximum of one step counter indicator (Digital Detection System) per bank inoperable either:

   a. Verify that all rod position indicators (Analog Detection System) for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 35 steps of each other at least once per 8 hours, or

   b. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

3. With more than one rod position indicator (Analog Detection System) per bank inoperable or more than one step counter indicator (Digital Detection System) per bank inoperable be in HOT STANDBY within 6 hours.

Basis: Control rod position and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per shift with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The indicator inoperability allowance of Specification C requires indirect measurement of rod position or a restriction in THERMAL POWERS; either of these restrictions provide assurance of fuel rod integrity during continued operation.

DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS
PROPOSED CHANGE NO. 93 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATIONAL LICENSE DPR-13

This is a request to (1) revise Appendix A Technical Specification 1.0, DEFINITIONS, and (2) add a new Section 3.0, LIMITING CONDITIONS FOR OPERATION (GENERAL).

Reason for Proposed Change

In an effort to clarify the meaning of the term Operable when used within the context of the Limiting Conditions for Operation, the NRC transmitted a request, by letter dated April 10, 1980, for submittal of proposed changes to the Technical Specifications which incorporate the requirements of the Model Technical Specifications included as Enclosure 1 to their letter. By letter dated May 15, 1980, a commitment was provided to the NRC for submittal of the requested proposed changes to the Technical Specifications.

Existing Specification

Technical Specification 1.0 currently includes the following definition of the term Operable:

"Operable:

Operable means that the system or component is completely capable of performing its required function in its required manner."

Technical Specification 3.0 does not currently exist.

Proposed Specification

Technical Specification 1.0 would be revised to include the following definition of the term Operable:

"Operable:

A system, subsystem, train, component or device shall be Operable or have Operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s)."

A new Section 3.0 will be added to the Limiting Conditions for Operation to read:

"3.0 LIMITING CONDITIONS FOR OPERATION (GENERAL)

Applicability: Applies to the operational requirements to be implemented when specific actions are not identified within individual Limiting Conditions for Operation.

Objective: To ensure that the station is placed in a safe condition when circumstances arise which are not identified within individual Limiting Conditions for Operation.

Specification: A. In the event a Limiting Condition for Operation and/or associated Action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least Hot Standby within 1 hour, and in at least Cold Shutdown within the following 30 hours unless corrective measures are completed that permit operation under the permissible Action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a mode of operation in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

B. When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered Operable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is Operable, and (2) all of its redundant system(s), subsystems(s), train(s), component(s), and device(s) are Operable, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least Hot Standby within 1 hour, and in at least Cold Shutdown within the following 30 hours. This specification is not applicable during the Cold Shutdown or Refueling modes of operation.

Basis: Specification A delineates the action to be taken for circumstances not directly provided for in the Action statements and whose occurrence would violate the intent of the specification. For example, Technical Specification 3.3 requires in part that two recirculation pumps be

Operable in order for the reactor to be made or maintained critical and provides explicit action requirements if one recirculation pump is inoperable. Under the terms of Specification A, if more than one recirculation pump is inoperable, the unit is required to be in at least Hot Standby within 1 hour and in at least Cold Shutdown within 30 hours unless corrective measures are completed. It is assumed that the unit is brought to the required mode of operation within the required times by promptly initiating and carrying out the appropriate action statement.

Specification B delineates what additional conditions must be satisfied to permit operation to continue, consistent with the Action statements for power sources, when a normal or emergency power source is not Operable. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component, or device in another division is inoperable for another reason.

The provisions of this specification permit the Action statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the Action statements of the associated electrical power source. It allows operation to be governed by the time limits of the Action statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual Action statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specfication 3.7 requires that two emergency diesel generators be Operable. The Action statement provides for a 72 hour out-of-service time when one emergency diesel generator is not Operable. If the definition of Operable were applied without consideration of Specification B, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification B permit the time limits for continued operation to be consistent with the Action statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be Operable, and all redundant systems, subsystems, trains, components and devices must be Operable, or otherwise satisfy Specification B (i.e., be capable of performing their design function and have at least one normal or one emergency power source Operable). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.7 requires in part that two physically independent offsite power lines be Operable. The Action statement provides a 24 hour out-of-service time when both required offsite power lines are not Operable. If the definition of Operable were applied without consideration of Specification B, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite power lines, would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable LCOs. However, the provisions of Specification B permit the time limits for continued operation to be consistent the the Action statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division, the emergency power source must be Operable (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be Operable, or likewise satisfy Specification B (i.e., be capable of performing their design functions and have an emergency power source Operable). In other words, both emergency power sources must be Operable and all redundant systems, subsystems, trains, components and devices in both divisions must also be Operable. If these conditions are not satisfied, shutdown is required in accordance with this specification.

In the Cold Shutdown or Refueling modes of operation, Specification B is not applicable, and thus the individual Action statements for each applicable Limiting Condition for Operation in these modes of operation must be adhered to."

Safety Analysis

The Technical Specifications are formulated to preserve the single failure criterion for systems that are relied upon in the Final Safety Analysis (FSA). By and large, the single failure criterion is preserved by specifying Limiting Conditions for Operation (LCOs) that require all redundant components of safety related systems to be Operable. When the required redundancy is not maintained, either due to equipment failure or maintenance outage, action is required, within a specified time, to change the operating mode of the plant to place it in a safe condition. The specified time to take action, usually called the equipment out-of-service time, is a temporary relaxation of the single failure criterion, which, consistent with overall system reliability considerations, provides a limited time to fix equipment or otherwise make it Operable. If equipment can be returned to Operable status within the specified time, plant shutdown is not required.

LCOs are specified for each safety related system in the plant, and with few exceptions, the Action statements address single outages of components, trains or subsystems. For any particular system, the LCO does not address multiple outages of redundant components, nor does it address the effects of outages of any suppport systems - such as electrical power or cooling water - that are relied upon to maintain the Operability of the particular system. This is because of the large number of combinations of these types of outages that are possible. Instead, the Technical Specification employ general specifications and an explicit definition of the term Operable to encompass all such cases. These provisions have been formulted to assure that no set of equipment outages would be allowed to persist that would result in the facility being in an unprotected condition. These specifications are contained in the Standard Technical Specifications and are incorporated into the San Onofre Unit 1 Technical Specifications by this Proposed Change. Illustrative examples of how these specifications apply are contained in the associated Bases.

Based upon the analysis provided 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 hazards 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.