DESCRIPTION OF PROPOSED CHANGES
NPF-10-50 AND NPF-15-50, REVISION 2 25
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.9.6, REFUELING MACHINE.

Existing Specifications

Unit 2: See Attachment "A"
Unit 3: See Attachment "B"

Proposed Specifications

Units 2 and 3: See Attachment "C"

Description

The proposed change would revise Technical Specification 3/4.9.6, Refueling Machine. Specification 3/4.9.6 delineates operability and surveillance requirements for the refueling machine to ensure that (1) the refueling machine will be used for movements of fuel assemblies and Control Element Assemblies (CEA's); (2) the refueling machine has sufficient load capacity to lift a fuel assembly; and (3) the core internals and pressure vessel are protected from excessive lifting forces in the event that they are inadvertently engaged during fuel handling operations. During refueling, the Control Element Drive Mechanism (CEDM) extension shaft are uncoupled and recoupled. Coupling and uncoupling is verified by weighing the CEA's and/or extension shafts. Both of these operations involve small movements of the CEA's. The NRC staff has interpreted that these small movements of CEA's are within the applicability of Specification 3/4.9.6. However, the refueling machine cannot be used for either coupling/uncoupling or weighing of CEA's. The proposed change would add a note which exempts use of the refueling machine for coupling/uncoupling and verification of coupling/uncoupling of CEA's.

Safety Analysis

The proposed changes discussed shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

FSAR Section 9.1.4.2.3.3, Refueling Procedure, describes the process for coupling and uncoupling the CEDM drive shaft extensions from their CEA's. FSAR Section 15.7.3.4.1, Design Basis Fuel Handling Accidents, describes the weighing of the CEA drive shafts to ensure uncoupling from the CEA's. This change is in accordance with the FSAR and there is no increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   As discussed above, this change is in accordance with FSAR Sections 9.1.4.2.3.3 and 15.7.3.4.1 and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   This change is in accordance with FSAR Sections 9.1.4.2.3.3 and 15.7.3.4.1 and does not involve a reduction in a margin of safety.

The Commission has provided guidance for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously-analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

SRP Section 9.1.4 discusses acceptance criteria for the fuel handling system. The objectives of the SRP are to preclude criticality accidents, and releases of radioactivity. Criticality accidents are, in part, prevented by verification of uncoupling of the CEA extension shafts prior to removal of the upper guide structure, thereby preventing CEA withdrawal when the upper guide structure is removed. The proposed change would permit coupling/uncoupling and verification of uncoupling, therefore reducing the probability of accidental criticality.

The proposed specification maintains the requirement to use the refueling machine for movements of fuel, requires the refueling machine to have sufficient capacity to lift a fuel assembly and requires an overload cutoff to assure that excessive forces are not applied. Therefore, the proposed change meets the SRP acceptance criteria and is similar to example (vi).

SAFETY AND SIGNIFICANT HAZARDS DETERMINATION

Based on the above Safety Analysis, it is concluded that; (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT A

(Existing Specification)

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

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* Except four finger CEAs.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other 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 the activity release assumed in the accident analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling train 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 the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop 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 the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

ATTACHMENT B

(Existing Specification)

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

*Except four finger CEAs.

NOV 15 1982

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other 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 the activity release assumed in the accident analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling train 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 the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating shutdown cooling loop 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 the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

NOV 15 1982

ATTACHMENT C
(Proposed Specifications)

REFUELING OPERATIONS

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

* Except for movement of four finger CEA's, coupling and uncoupling the CEA extension shafts or verifying the coupling and uncoupling.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

Coupling and uncoupling of the CEA's and the CEA extension shafts is accomplished using the gripper operating tool. The coupling and uncoupling is verified by weighing the CEA's and/or extension shafts.

3/4.9.7 FUEL HANDING MACHINE - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other 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 the activity release assumed in the accident analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

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

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating shutdown cooling loop 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 the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-52 AND SAFETY EVALUATION

This is a request to revise Technical Specifications 3/4.2.4 and 3.3.1 (Table 3.3-1, Action 6) and associated bases.

Existing Specification

See Attachment A for Units 2 and 3

Proposed Specification

See Attachment B for Units 2 and 3

Reason for Proposed Change

The purpose of this proposed change is to 1) improve DNBR operating margin when COLSS is out-of-service, 2) allow continued operation beyond a 7 day period when both CEA calculators (CEAC's) are out-of-service, and 3) clarify Action Statement 6 of Table 3.3-1 when COLSS is in or out-of-service.

Description

DNBR Operating Margin Improvement

Description

The CPC and COLSS systems are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the event of an Anticipated Operational Occurrence (AOO), the CPC's will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow transient which is the limiting AOO. When the COLSS is Out of Service (COOS) the monitoring function is performed via the CPC calculation of DNBR in conjunction with a COOS Limit Line (Technical Specification Figure 3.2-2) which restricts the reactor power sufficiently to preserve the ROPM.

This proposed change increases the COLSS out-of-service margin by 10%. The thermal margin gain is reflected in the new COLSS out-of-service limit lines as presented in revised Figure 3.2-2 and new Figure 3.2-3 of Attachment B. These lines take into account a reevaluated CPC overall uncertainty consistent with the requirements of LCO monitoring.

Note that two COLSS out-of-service curves are being proposed: one applicable at and above 80% rated thermal power and one applicable below 80% rated thermal power. This is done to maximize allowable power level near full power operation.

Improvement If COLSS and/or CEAC's Are Out-of-Service

Normally, the Control Element Assembly Calculators (CEACs) assist the Core Protection Calculators (CPCs) in providing protection against various transients involving CEA deviations. This is done by providing CPCs with Departure from Nucleate Boiling Ratio (DNBR) and Linear Heat Rate (LHR) penalties based on the effect the deviation has on the radial peaking factor. Rod position information is provided by the Reed Switch Position Transmitters (RSPT) to the two CEACs.

When both CEACs are "out-of-sevice", the CEACs can no longer be relied upon to provide appropriate penalties to the CPCs. One of the actions required by the Technical Specifications (Table 3.3-1, Action Item 6) is to set the RSPT/CEAC addressable constant to the inoperable status. This action causes the CPC to ignore all CEA position information and CEAC penalty factors. Instead, the CPC uses off-line calculated penalty factors. The purpose of these penalties is to compensate for the lack of CEA position information or CEAC penalties for certain events (e.g., CEA withdrawal) from the limited CEA insertion permitted by Action Item 6.

For example, if a single CEA is withdrawn from an initially inserted group, this will result in an increase in radial peak. However, since the CPCs are no longer receiving penalties from the CEACs, the off-line calculated penalties are needed to assure the fuel design limits on DNBR and LHR are not violated. The off-line calculated penalties are continuously applied since, due to the lack of CEA position information, the CPCs would not know if a CEA withdrawal were to occur. Note that all other parameter changes (e.g., inlet temperature, pressure, etc.) will be correctly "seen" by the CPCs. Thus, the penalties need only account for the effect of a possibly withdrawn CEA or CEA group.

For CEA drop events (single or subgroup), the CPCs will not detect the deviation. The other parameters monitored by CPC will indicate an increase in DNBR and decrease in LHR (i.e., power and T_c will decrease). Thus, a CPC trip will not occur. As a result, sufficient initial margin must be required to allow the event to occur without violating the fuel design limits. Since the Technical Specifications allow one hour following a CEA drop before action is required, the effects of xenon on radial peak for one hour following a drop must be included. The CEA drop events become the limiting events - requiring more thermal margin than the four pump loss of flow (the limiting event when CEACs are in-service).

The COLSS assures that enough thermal margin is available so that the four pump loss of flow does not violate the DNBR fuel design limit. Since more margin is required to protect against CEA drops when both CEAC's are out-of-service, an additional penalty on the COLSS calculated power operating limit is needed. This is the reason for the 22% penalty incorporated into Figure 3.2-1 for both CEAC's out-of-service.

Naturally, if COLSS is unavailable, the "monitoring" of the required thermal margin must be done using the CPCs. As stated previously, CPCs will use an off-line calcualted penalty factor to calculate DNBR when CEACs are out-of-service. This penalty, which was generated to protect against CEA withdrawal events, can be credited for CEA drop events. In this case, an additional penalty of 1.13 on BERR1 is needed to assure sufficient margin is available to protect against the CEA drop events. In other words, the combination of the off-line calculated CEA withdrawal penalty and the 1.13 factor on CPC assures that the same initial margin is available as would be when the COLSS calculated power operating limit is reduced by 22%. Both are ways of assuring CEA drops will not violate fuel design limits.

Table 1 outlines the actions and approximate power levels attainable under the current Technical Specifications.

TABLE 1

COLSS/CEAC's OUT OF SERVICE (O.O.S) ACTIONS
TO ASSURE NECESSARY THERMAL MARGIN

	COLSS(1) O.O.S.	CEAC's(2)(3) O.O.S.	COLSS and CEAC(3) O.O.S.
Tech. Spec. Action	Use DNBR Curve from Tech. Spec. 3.2.4	Reduce COLSS Calculated POL by 22% of full power	Apply 1.13 Multiplier on BERR1
CPC Penalty Automatically Applied	No	Yes(4)	Yes(4)
Approximate Power Capability	85%	70-75%	50-60%

(1) At least one CEAC in-service

(2) COLSS in-service

(3) When CEAC's are both O.O.S.; RSPT/CEAC Inoperable constant is set, CEA insertion is restricted and CEDMCS is placed in "OFF". (See action item 6 to table 3.3-1).

(4) Setting the inoperable constant automatically applies a penalty to CPC DNBR calculation.

Safety Evaluation

The proposed changes discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The probability or consequences of an accident previously evaluated will not be significantly increased. The proposed changes affect the margin of safety of accidents previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The dynamic uncertainty components included in the CPC DNBR calculation, but not required in the LCO monitoring system, are removed from the COOS DNBR limit line. Previously, with COOS, the CPC's provided the monitoring function on DNBR very conservatively using the larger safety system uncertainties.

   The conservatisms in the CPC uncertainty analysis must cover many transient conditions and a very wide operating space. By limiting these requirements to be consistent with LCO monitoring, the uncertainty factors which are required can be reduced. The margin corresponding to this reduction in uncertainty factors is used in providing a new, less restrictive COLSS out-of-service limit. Note that none of the CPC channels is affected and, therefore, the safety system setpoints are unchanged. The margin credit for the reduced uncertainties is used to determine a less restrictive COLSS out-of-service limit.

   If one CEAC is inoperable for beyond the 7 day period, plant safety will not be compromised because both CEAC's will be flagged inoperable to the CPC's activating default penalties which will preserve the safe operating margin.

The proposed change is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

Based on the Safety Evaluation, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92 and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which signficantly alters the impact of the station on the environment as described in the NRC Environmental Statement.

GPvN:1D39F

ATTACHMENT A

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

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SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

Figure 3.2-1

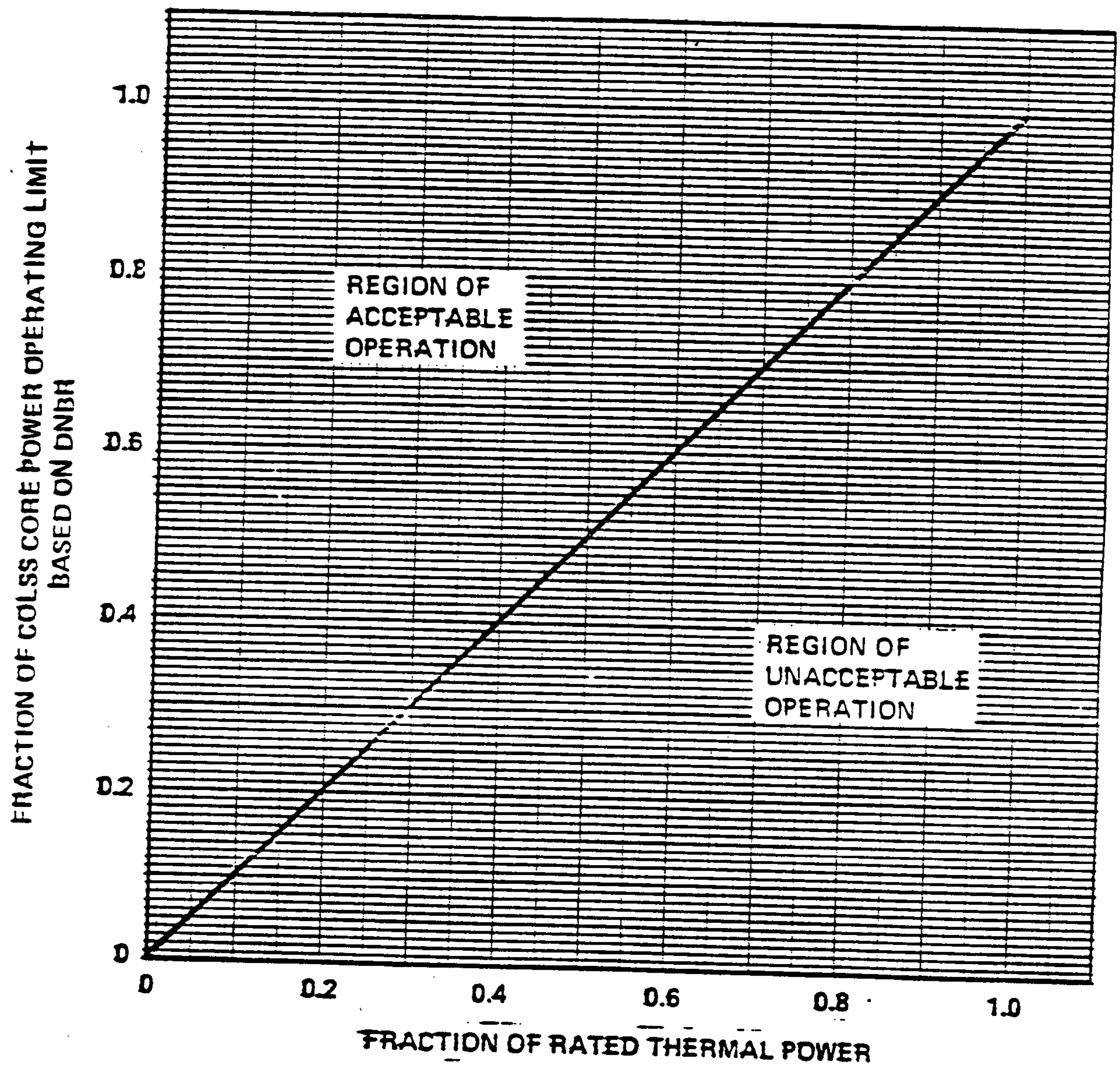

Figure 3.2-1 DNBR margin operating limit based on COLSS.

Figure 3.2-2

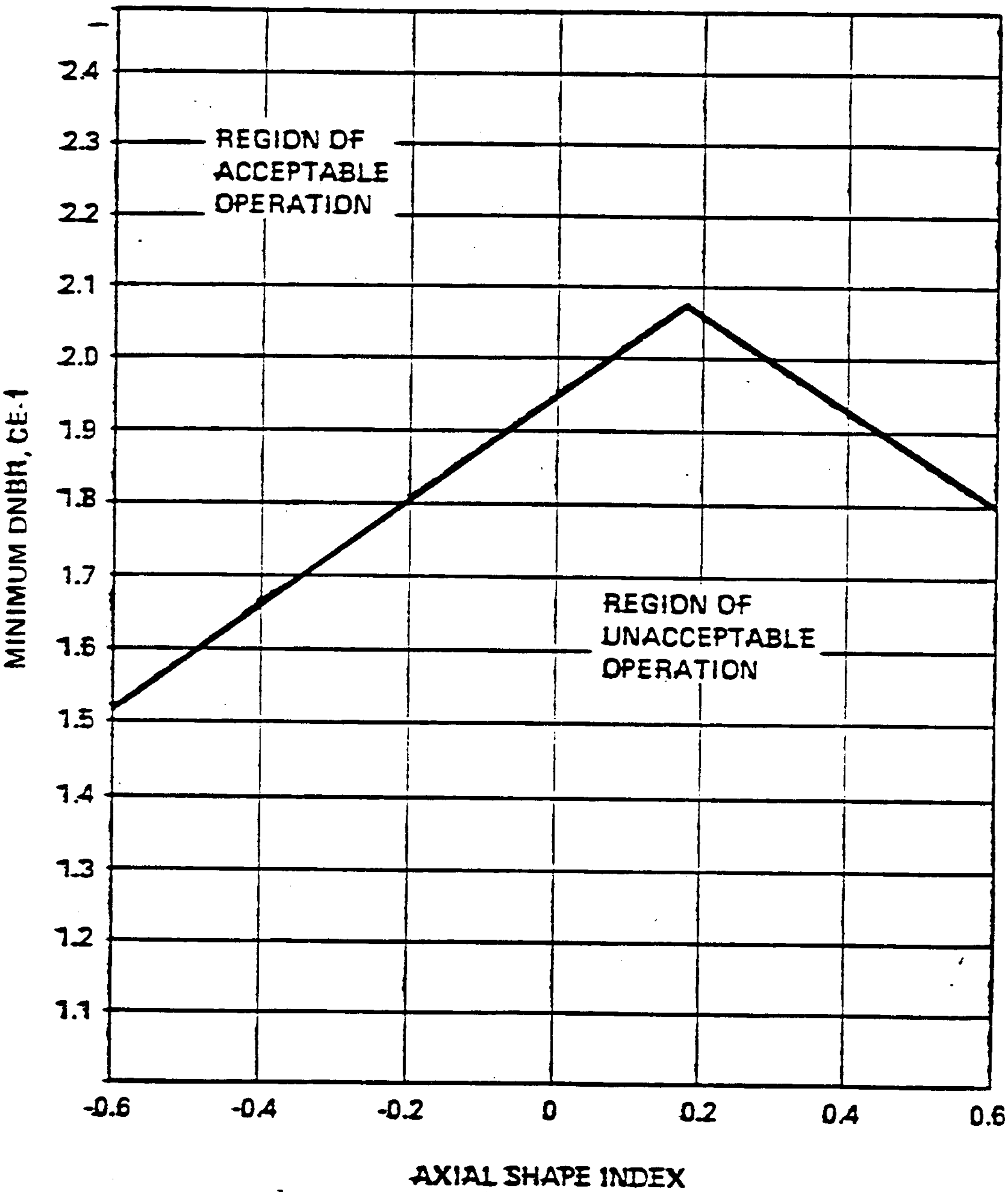

Figure 3.2-2 DNBR margin operating limit based on core protection calculators. (COLSS out of service)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and

TABLE NOTATION

maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER.

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 1. a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 7 — With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A — With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

NPF-10-52
NPF-15-52

ATTACHMENT B

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable. change to: 3.2-1, 3.2-2, or 3.2-3

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2. Add: or Figure 3.2-3, whichever is applicable.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

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SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

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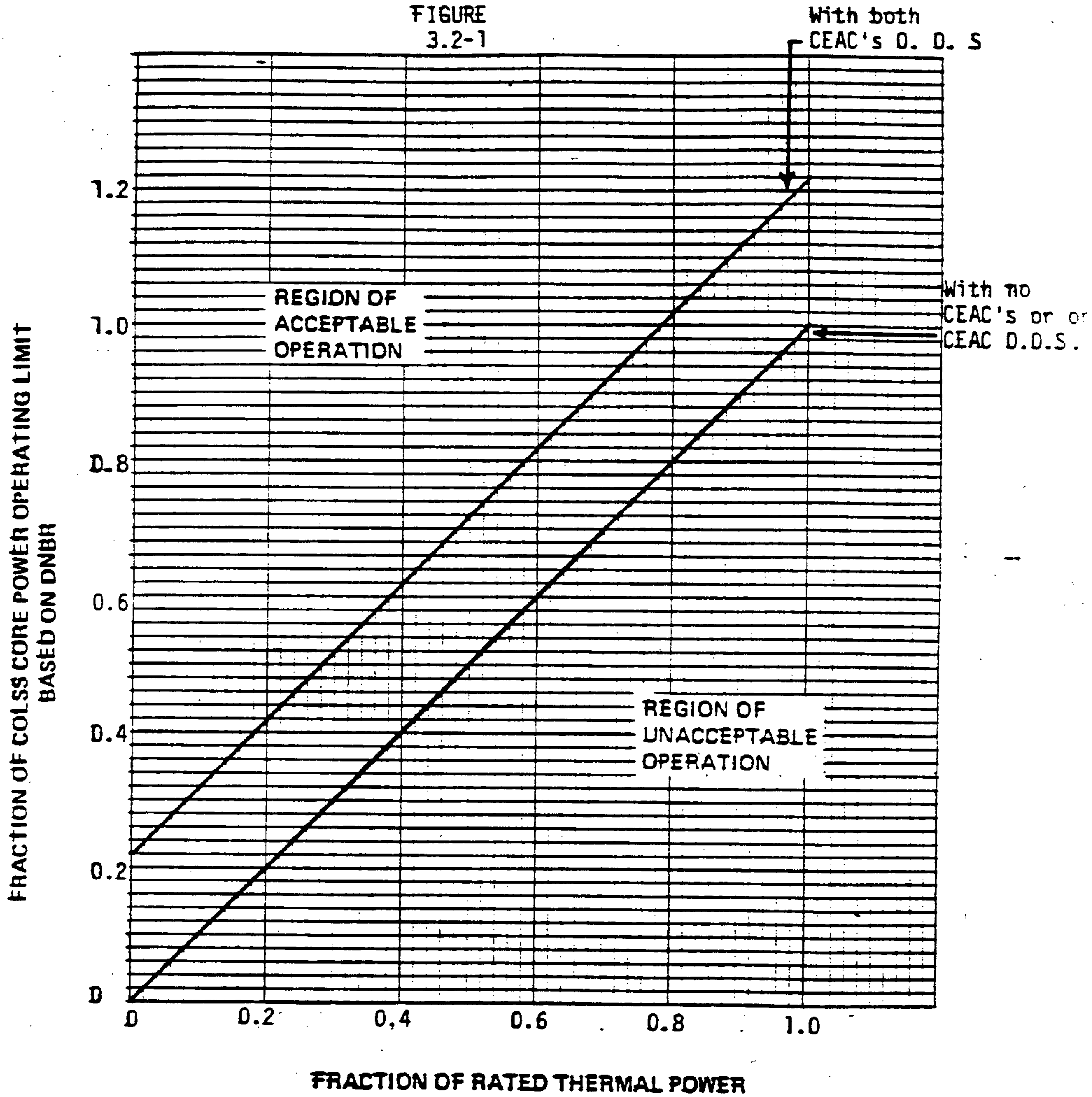
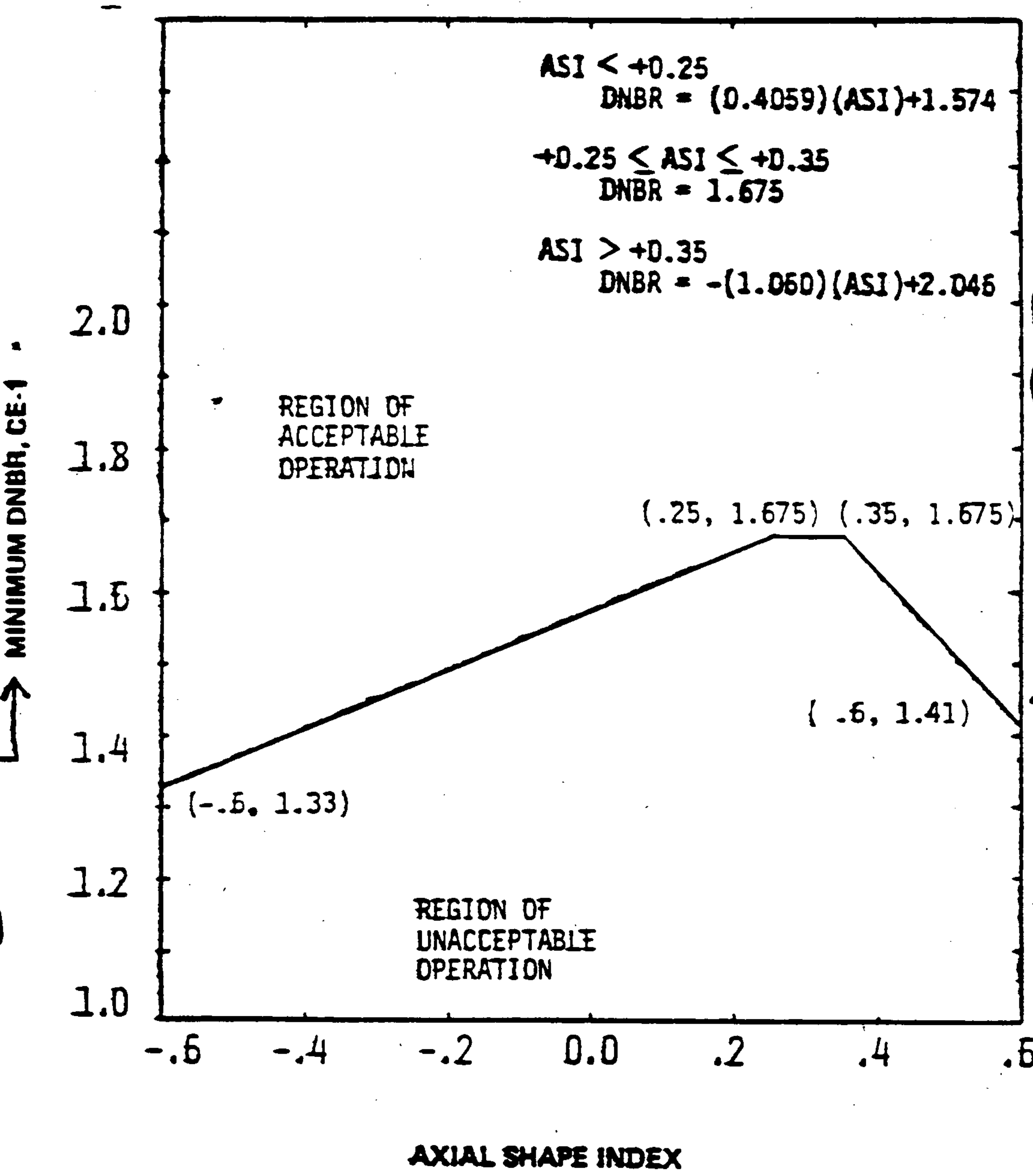


~~Figure 3.2-1~~ DNBR ~~margin operating limit based on COLSS.~~

Figure 3.2-2

Add: CPC



Revised curve

Figure 3.2-2 DNBR margin operating limit based on core protection calculators. (COLSS out of service)

Add: with RATED THERMAL POWER ≥ 80%.

Change to: 3.2-3

New Page

Figure 3.2-2

Add: CPC

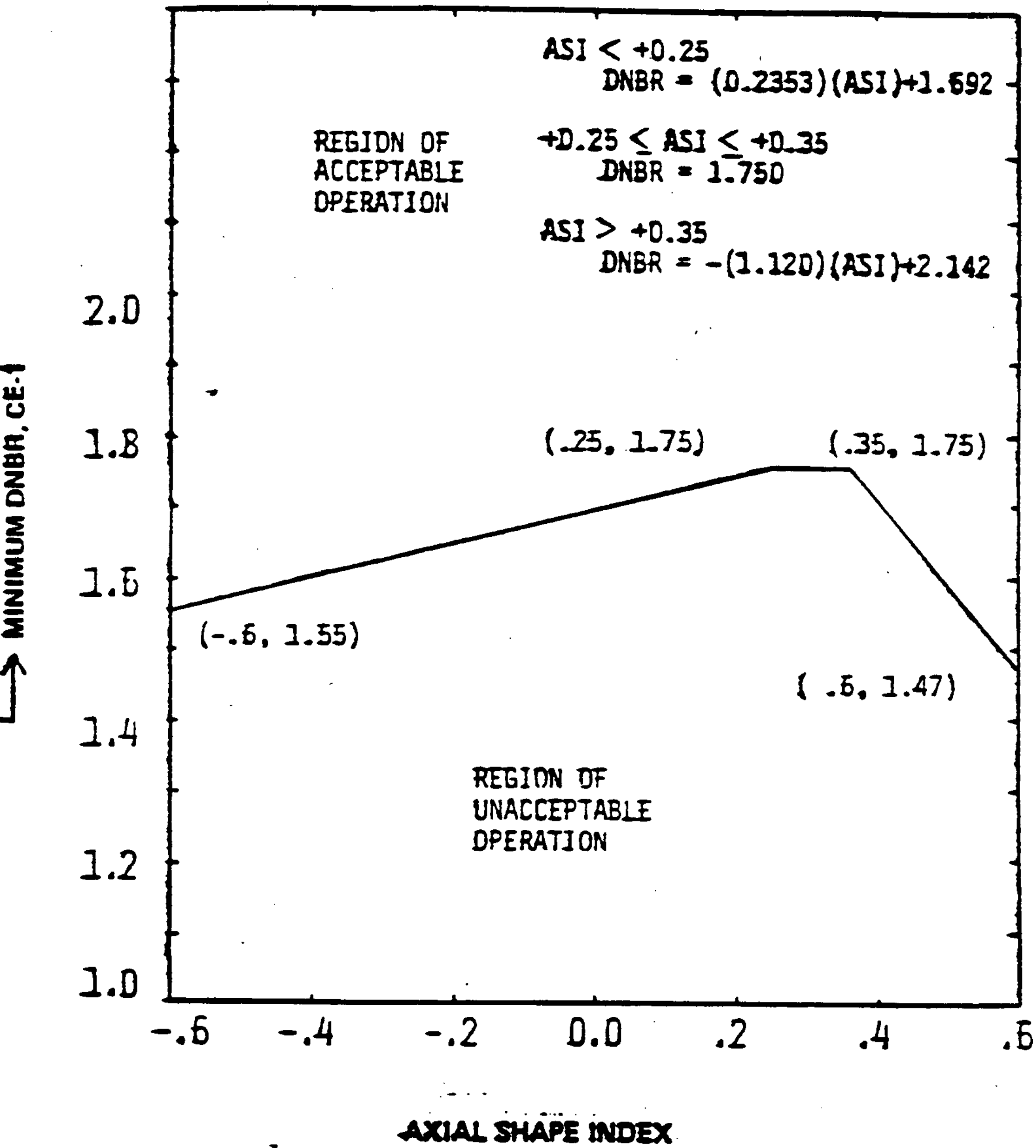


Figure 3.2-2 DNBR margin operating limit ~~based on core protection calculators~~ (COLSS out of service)

Change to: 3.2-3

Add: with RATED THERMAL POWER < 80%.

REVISION TO ACTION ITEM 6 OF TABLE 3.3-1 IN SPECIFICATION 3.3.1

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that Action Items 6.b.1, .2 and .3 are met with COLSS is in-service, or Action Items 6.c.1, .2 and .3 are met with COLSS out-of-service*.

b. With both CEAC's inoperable and COLSS in-service, operation may continue provided that:*

1. Within 1 hour the DNBR margin operating limit required by Specification 3.2.4 (Figure 3.2-1) is satisfied for both CEAC's out-of-service.

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

c. With both CEAC's inoperable and COLSS out-of-service operation may continue provided that:*

1. Within 1 hour multiply the CPC value of BERR1 corresponding to COLSS in-service by 1.13 and re-enter into the CPC's.

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all othr CEA's in its group.

---

* Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

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MOVABLE CONTROL ASSEMBLIES (Continued)

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours 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 maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

ADD

Setting the "RSPT/CEAC Inoperable" addressable constant in the CPC's to indicate to the CPC's that one or both of the CEAC's is inoperable does not necessarily constitute the inoperability of the RSPT rod indications from the respective CEAC. Operability of the CEAC rod indications is determined from the normal surveillance

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 Add: or 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Insert A

A: Uncertainty terms already taken into account ~~required only~~ in the CPC's safety monitoring are removed from Figures 3.2-2 and 3.2-3 since the curves are intended to monitor only the LCO during steady state operation.

Add A to last line on page B 3/4 2-3

DESCRIPTION OF PROPOSED CHANGES NPF-10-141 AND NPF-15-141 AND SAFETY ANALYSIS

This is a request to revise Section 4.4.4.4 (steam generator) Acceptance Criteria of the Technical Specifications for San Onofre Nuclear Generating Station Units 2 and 3.

Description

Technical Specification 3/4.4.4 requires steam generator operability and specifies Surveillance Requirements to verify steam generator integrity. the current acceptable level of steam generator tube wall thinning shown in Figure 4.4.1, is 44% for tube rows 0 through 92 and decreases linearly to 26% in tube row 147. The proposed change will delete Figure 4.4.1 and specify a tube thinning limit of 44% for all steam generator tubes.

The original analysis established the structural adequacy of the San Onofre Units 2 and 3 steam generator tubes and tube supports, when subjected to various hypothetical accident conditions. It was determined that the limiting event was a combination of a Loss of Coolant Accident (LOCA) and Safe Shutdown Earthquake (SSE). The calculated stresses ocurring in the steam generator tube walls as a result of the limiting event were compared to the maximum allowable stresses as defined by the NRC staff's criteria (Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes). The analysis indicated that degradation of up to 64 percent is acceptable for both the straight portion of all tubes and the "U" bend region for the majority of tube rows. The outer tube rows experienced significant stress in the "U" tube bend region, due to the combination of hydraulic loads associated with blowdown of the primary system as a result of the LOCA and earthquake-induced accelerations resulting from the SSE. Thus in the outer tube bundle bend regions, the allowable degradation decreased linearly from 64 percent to a minumum of 46 percent at the outermost row.

The proposed change will remove excessive conservatism from the assumptions used in the original analysis and thereby establish a more accurate steam generator tube thinning limit. In addition, the proposed change prevents unnecessary (1) plugging of tubes, (2) associated high personnel radiation exposure and (3) decreases in the steam generator heat transfer surface area.

The revised analysis of the limiting LOCA/SSE scenario credits the frictional or binding restraint on the tubes provided by the vertical tube supports in the horizontal tube run on top of the "U" tube span; this was previously neglected in the original calculations. In addition, the LOCA and SSE peak loads were combined by square-root-of-the-sum-of-the-squares (SRSS) combination; these loads were added in the original calculations. The combination of a LOCA and SSE is still the limiting event. The revised analysis shows that tube degradation of up to 64 percent is acceptable for all steam generator tubes in meeting the criteria of Regulatory Guide 1.121.

In establishing the Technical Specification limits for San Onofre Units 2 and 3, the NRC imposed a 20 percent reduction in the above allowable degradation to cover Eddy Current Testing (ECT) measurement error and continued degradation through the next fuel cycle (in response to guidance provided in Regulatory Guide 1.121). As a result, a steam generator tube degradation limit of 44 percent for all steam generator tubes is proposed.

Existing Technical Specifications

Unit 2

See Attachment A

Unit 3

See Attachment B

Proposed Technical Specifications

Unit 2

See Attachment C

Unit 3

See Attachment D

Safety Analysis

The proposed changes discussed above shall be deemed to constitute a signficant hazards consideration if there is a positive finding in any of the following areas.

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed Steam generator tube wall thinning acceptance criteria has been determined in accordance with Regulatory Guide 1.121. This criteria has been established to provide reasonable assurance that tube failure will not occur during operation of the plant; particularly during a LOCA, SSE or Main Steam Line Break. Removal of excessive conservatism from the steam generator tube stress analysis does not violate the Regulatory Guide 1.121 criteria and will not produce conditions which could lead to steam generator tube rupture under the most limiting design basis accident conditons.

2. Will operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

The proposed change does not change the configuration of the plant or the way in which it is operated. Therefore, the change does not create the possibility for a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change affects the tube wall thinning acceptance criteria for tube rows beyond row 92. The supporting steam generator tube stress analysis meets the criteria of Regulatory Guide 1.121. Therefore the proposed change does not result in a significant reduction of a margin of safety.

The Commission has provided guidance concerning the application of starndards for determining whether a significant hazards consideration exists by providing certain examples of amendments (48 FR 14870) that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously-analyzed accident or may in some way reduce a safety margin, but where the results of the proposed change are clearly within all acceptance criteria for the system or component specified in the Standard Review Plan (SRP). Section 5.4.2.2 of the SRP references Regulatory Guide 1.83 which specifies inservice inspection criteria for determining steam generator operability. The proposed change prescribes steam generator tube thinning criteria which was developed in accordance with Regulatory Guide 1.83 and the SRP.

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of any accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to the nominal tube wall thickness as determined in Figure 4.4-1.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

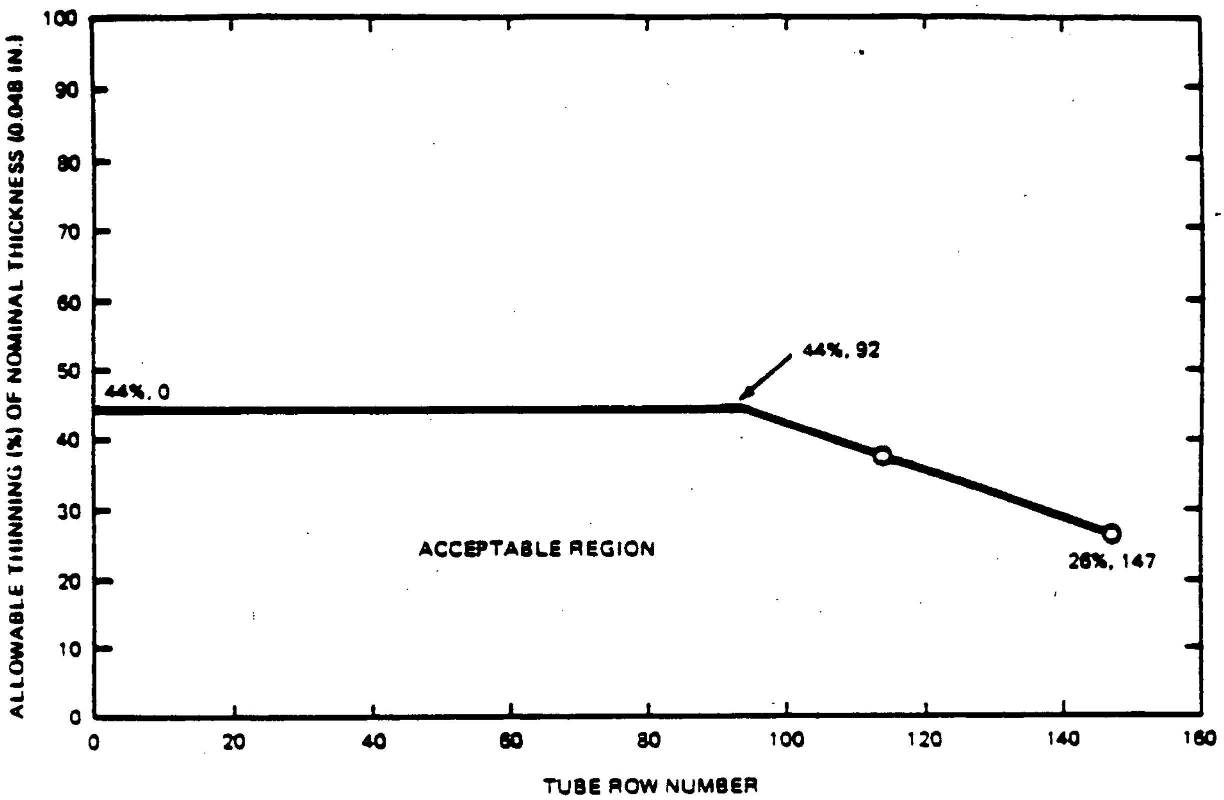

Figure 4.4-1
TUBE WALL THINNING ACCEPTANCE CRITERIA

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of Figure 4.4-1. Figure 4.4-1 was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. These results are controlling items in allowable tube wall thinning for tube rows 92 to 147. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

ATTACHMENT B

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to the nominal tube wall thickness as determined in Figure 4.4-1.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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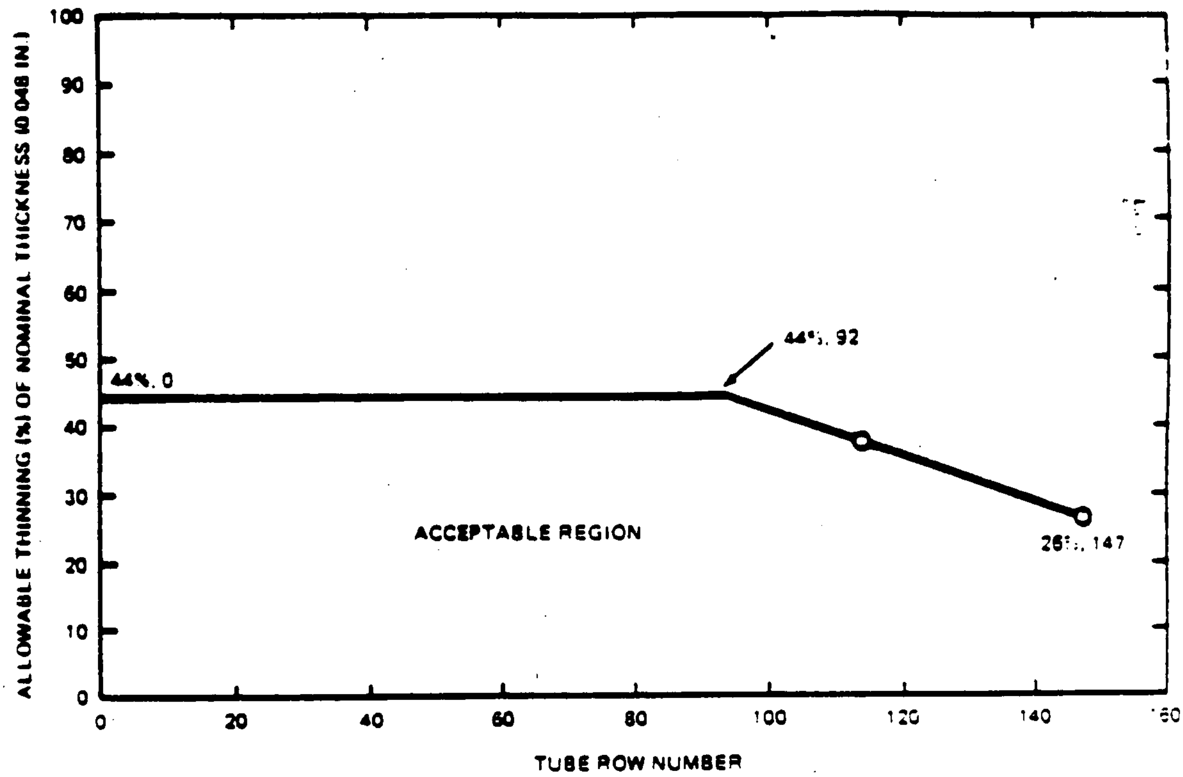

Figure 4.4-1
TUBE WALL THINNING ACCEPTANCE CRITERIA

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STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of Figure 4.4-1. Figure 4.4-1 was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. These results are controlling items in allowable tube wall thinning for tube rows 92 to 147. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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ATTACHMENT C

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occuring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 44% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding 44% of the nominal tube wall thickness. This criteria was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

ATTACHMENT D

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occuring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 44% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding 44% of the nominal tube wall thickness. This criteria was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

DESCRIPTION OF PROPOSED CHANGE NPF-10-126
AND SAFETY ANALYSIS

This is a request to revise Section 3/4.5.2 ECCS Subsystems - Tavg Greater Than or Equal to 350°F of the Technical Specifications for San Onofre Nuclear Generation Station Unit 2.

Description

The proposed change involves a revision to Surveillance Requirement 4.5.2.a, which specifies valve functions and positions required for Emergency Core Cooling System operability.

The proposed amendment would revise Technical Specification 3/4.5.2, ECCS Subsystems - Tavg Greater Than or Equal to 350°F. Section 3/4.5.2 requires Emergency Core Cooling System (ECCS) operability and specifies Surveillance Requirements to verify such operability. Surveillance Requirement 4.5.2.a specifies valve positions required for ECCS subsystem operability. The proposed change would revise Section 4.5.2.a to be consistent with modifications made to the Shutdown Cooling System (SDCS) in accordance with NRC Branch Technical Position RSB 5-1. The SDCS modifications will provide remote alignment capability from the control room. Previously manual valve pre-alignment was required prior to SDCS operation.

Specifically, the proposed change to Section 4.5.2.a includes the addition of two new SDCS bypass flow control valves (HV 8160, HV 8161) and Low Pressure Safety Injection (LPSI) Pump miniflow isolation valves (HV 8162 and HV 8163); replacement of the existing SDCS flow control valve (FV 0306 replaced by HV 0396); and deletion of the SDCS heat exchanger flow control valve and isolation valves (HV 9316, 14-78 and 14-80), SDCS bypass flow control/isolation valve 14-153 and isolation valves 14-81 and 14-82. The proposed change to Section 4.5.2.a will verify the correct valve alignment for ECCS subsystem operability following completion of the SDCS design modification.

The new SDCS bypass flow control valves (HV 8160 and HV 8161) provide for redundant, remotely operable, Class 1E bypass flow control. HV 0396 and HV 8161 are powered by the opposite train from HV 8160 consistent with single failure design criteria. In the event that power to HV 8160 (normally used for flow control) is lost, HV 8161 will be closed and HV 0396 used to provide the required bypass flow control. HV 0396, HV 8160 and HV 8161 replace FV 0306 and 14-153 to provide remote operation capability, consistent with BTP RSB 5-1. The existing non Class 1E-powered SDCS heat exchanger flow control valve and associated isolation valves (HV 9316, 14-78 and 14-80) are to be removed and the flow control function performed by new valves HV 8150 and HV 8151 (redundant, remotely operable and Class 1E powered).

Motor-operated LPSI miniflow isolation valves HV 8162 and HV 8163 are to be added to provide remote isolation capability consistent with BTP 5-1. Isolation of the miniflow lines is required to prevent transport of potentially contaminated primary coolant to the Refueling Water Storage Tank (RWST). The valves will be powered from the opposite train as the associated LPSI pump to prevent the loss of a train of emergency power from resulting in a potentially uncontrolled flow path from the Reactor Coolant System to the RWST.

Isolation valves 14-81 and 14-82 are to be removed from Section 4.5.2.a. The closure of 14-81 and 14-82 [isolation valves for HV 0396 (normally closed)] has been previously analyzed for this configuration in the FSAR failure modes and effects analysis of the Unit 3 Safety Injection System (FSAR Table 6.3-1 for Unit 3, Item 14). It was concluded that inadvertent closure of these valves would have no effect on ECCS operation, since HV 8160 (open) and HV 8161 (open) bypass 14-81 and 14-82 and provide the normal ECCS flowpath. In addition, surveillance of these valves requires frequent entry into a confined contaminated area (14-81 and 14-82 are not equipped with remote position indication) with associated personnel radiation exposure.

The SDCS design change (DCP 29N) has been reviewed and approved and does not involve an unresolved safety issue. A similar design change was implemented at Unit 3 prior to initial plant startup. This proposed change is required following completion of the SDCS modifications and prior to entry into Mode 3 during the Unit 2 refueling outage.

Existing Technical Specification

See Attachment A

Proposed Technical Specification

See Attachment B

Safety Analysis

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probabiliy or consequences of an accident previously evaluated?

   Response: No

   The proposed change involves an administrative revision of Section 4.5.2.a consistent with the modifications to the SDCS implemented through DCP 29N (in accordance with BTP RSB 5-1), as described above. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   The proposed change and associated plant modifications (DCP 29N) are in accordance with BTP RSB 5-1 and do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Reponse: No

   DCP 29N has been reviewed and approved and the proposed modifications (described above) do not involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vii) relates to a change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations. San Onofre Nuclear Generating Station Unit 2 was originally designed to include manual operator action to reach cold shutdown conditions following a design basis accident, in keeping with the then existing criteria. However, with the advent of NRC Branch Technical Position (BTP) RSB 5-1, licensees were required to provide the capability to reach cold shutdown through operator action from the control room (as specified in Standard Reveiw Plan Section 5.4.7). NUREG-0712 Section 5.4.3, requires that the design modifications to provide this capability be implemented prior to startup following the first refueling outage for San Onofre Unit 2. Therefore, the proposed change is simiar to example (vii) in that it reflects compliance with a change in NRC regulations.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not involve a significant hazards consideration as defined by 10 CFR 50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	FV0306	SDC Bypass Flow Control	LOCKED OPEN (THROTTLED)(MANUAL)
h.	14-153		LOCKED CLOSED (MANUAL)
i.	14-081		LOCKED OPEN (MANUAL)
j.	14-082		LOCKED OPEN (MANUAL)
k.	HV9420	Hot Leg Injection Isolation	CLOSED
l.	HV9434	Hot Leg Injection Isolation	CLOSED
m.	HV9316	SDC(HX) Flow Control	OPEN (THROTTLED)(AIR REMOVED)
n.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
o.	14-78	HV9316 Isolation	LOCKED OPEN (MANUAL)
p.	14-80	HV9316 Isolation	LOCKED OPEN (MANUAL)

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

ATTACHMENT B

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	HV0396	SDC Bypass Flow Control	CLOSED
h.	HV8161	SDC(HX) Bypass Flow Isolation	OPEN
i.	HV9420	Hot Leg Injection Isolation	CLOSED
j.	HV9434	Hot Leg Injection Isolation	CLOSED
k.	HV8160	SDC Bypass Flow Control	OPEN
l.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
m.	HV8162	LPSI Miniflow Isolation	OPEN
n.	HV8163	LPSI Miniflow Isolation	OPEN

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. Veifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

DESCRIPTION OF PROPOSED CHANGES NPF-10-137 AND NPF-15-137 AND SAFETY ANALYSIS

This is a request to change Section 3/4.8.1.2 "AC Sources, Shutdown" of the Technical Specifications for San Onofre Nuclear Generating Station, Units 2 and 3.

Description

The Surveillance Requirements governing emergency diesel generator (EDG) operability in MODES 5 and 6 (Section 4.8.1.2) currently prescribe all those surveillances required in MODES 1 through 4 (by Specification 4.8.1.1) with one exception (Section 4.8.1.1.2.a.5). Sections 3/4.8.1.2 and B 3/4.8 have been revised to include only those Limiting Conditions for Operation and Surveillance Requirements which verify operability of the AC Sources required under SHUTDOWN and REFUELING conditions (Modes 5 and 6) as applicable.

The largest anticipated load, in MODES 5 and 6 (considering all loads required to mitigate the consequences of the range of postulated accidents and all loads which facilitate plant operation and maintenance), has been calculated to be less than 80% of the EDG full rated capacity. Section 3.8.1.2.b.2 has been revised to specify a fuel storage capacity (37,600 gal.) consistent with the above calculated load requirement, ANSI N195-1976 and Regulatory Guide 1.137.

Automatic start of the EDG on an ESF signal, on loss of offsite power in conjunction with an ESF signal or from a test mode is not required in MODES 5 and 6. Therefore, Sections 4.8.1.1.2.d.5, d.6 and d.10 have been deleted. Automatic load sequencing on an ESF signal is also not required in MODES 5 and 6 therefore Section 4.8.1.1.2.d.12 has been deleted. No loads except the permanently connected shutdown loads are automatically connected to the EDG in MODES 5 and 6, therefore Section 4.8.1.1.2.d.8, which specifies the maximum auto-connected loads applicable in Modes 1 through 4, has been deleted. The exception to Section 4.1.1.2.a.5, as originally specified, has been retained.

It should be noted that the section numbers of the Surveillance Requirements deleted above are consistent with, and assume NRC acceptance of, the revisions submitted in NPF-10-91 and NPF-15-91.

Existing Specifications

Unit 2: See Attachment A

Unit 3: See Attachment B

Proposed Specifications

Unit 2: See Attachment C

Unit 3: See Attachment D

Safety Analysis

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   As described above, the proposed change prescribes Surveillance Requirements and Limiting Conditions for Operation which are consistent with the EDG support function required in MODES 5 and 6.

   The required fuel storage capacity has been determined based on the largest anticipated EDG load in MODES 5 and 6 in accordance with Regulatory Guide 1.9. Surveillance Requirements which do not apply to the range of design basis events of MODES 5 and 6 have been deleted. The reliability of the EDG in performing its function in MODES 5 and 6 remains unaffected by this change. Therefore, the probability or consequences of an accident remain bounded by existing analyses.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not alter the system configuration, operation or support function in the applicable modes. The proposed change verifies operability of the EDG by specifying Limiting Conditions for Operation and Surveillance Requirments consistent with the maximum required load in Modes 5 and 6. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change does not diminish EDG reliability or operability with respect to it's required support function in MODES 5 and 6. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The proposed Technical Specification changes are representative of 48 FR 14870 Section 50.91, example (vi) dated April 6, 1983, Subject: Amendments That Are Considered Not Likely to Involve a Significant Hazards Consideration, in that the results of the proposed change are clearly within all acceptable criteria.

Safety and Significant Hazards Determination

Based on the Safety Analysis, it is concluded that: (1) the proposed change does not involve a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the pbulic will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

SB:0787F

ATTACHMENT A

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day fuel tanks containing a minimum volume of 325 gallons of fuel,

2. A fuel storage system containing a minimum volume of 47,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and movement of irradiated fuel, or operation of the fuel handling machine with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5) and 4.8.1.1.3.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

as well as operation of loss of voltage logic, is the same as for the primary connection using the reserve auxiliary transformer, with the exception of no transfer to the companion unit.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Reg. Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975. However, ASTM-D270-1965 was reverified in 1975 rather than re-issued. The reverified 1965 standard is therefore the appropriate standard to be used.

Additionally, Regulatory Guide 1.9 allows loading of the diesel generator to its 2000 hour rating in an accident situation. The full load, continuous operation rating for each diesel generator is 4700 kw, while the calculated accident loading is 4000 kw. No 2000 hour loading has been specified by the diesel generator manufacturer and, as a result the full loading rating of 4700 kw is conservatively established as the 2000 hour rating. Diesel frequency droop restrictions are established due to HPSI flow rate considerations.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

ATTACHMENT B

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day fuel tanks containing a minimum volume of 325 gallons of fuel,

2. A fuel storage system containing a minimum volume of 47,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and movement of irradiated fuel, or operation of the fuel handling machine with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5) and 4.8.1.1.3.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

as well as operation of loss of voltage logic, is the same as for the primary connection using the reserve auxiliary transformer, with the exception of no transfer to the companion unit.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Regulatory Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975. However, ASTM-D270-1965 was reverified in 1975 rather than re-issued. The reverified 1965 standard is therefore the appropriate standard to be used.

Additionally, Regulatory Guide 1.9 allows loading of the diesel generator to its 2000 hour rating in an accident situation. The full load, continuous operation rating for each diesel generator is 4700 kw, while the calculated accident loading is 4000 kw. No 2000 hour loading has been specified by the diesel generator manufacturer and, as a result the full loading rating of 4700 kw is conservatively established as the 2000 hour rating. Diesel frequency droop restrictions are established due to HPSI flow rate considerations.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Attachment C

ELECTRICAL POWER SYSTEMS

AC SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day fuel tanks containing a minimum volume of 325 gallons of fuel,

2. A fuel storage system containing a minimum volume of 37,600 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum AC electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and movement of irradiated fuel, or operation of the fuel handling machine with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required circuit between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at

SURVEILLANCE REQUIREMENTS (continued)

least once per 7 days by verifying correct breaker alignment and indicated power availability.

a. If the above required offsite source is supplied through the Unit 3 4160 volt Emergency Bus #3A04, the following buses are required:

   480 volt Emergency Bus #3B04
   125 volt Emergency Bus #3D1

b. If the above required offsite source is supplied through the Unit 3 4160 volt Emergency Bus #3A06, the following buses are required:

   480 volt Emergency Bus #3B06
   125 volt Emergency Bus #3D2

4.8.1.2.2 The above required diesel generator shall be demonstrated OPERABLE by performing the Surveillance Requirements of 4.8.1.1.2 (except 4.8.1.1.2 a.5, d.5, d.6, d.8, d.10 and d.12) and 4.8.1.1.3.

AC SOURCES, DC SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (continued)

as well as operation of loss of voltage logic, is the same as for the primary connection using the reserve auxiliary transformer, with the exception of no transfer to the companion unit.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Reg. Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975. However, ASTM-D270-1965 was reverified in 1975 rather than re-issued. The reverified 1965 standard is therefore the approproate standard to be used.

Additionally, Regulatory Guide 1.9 allows loading of the diesel generator to its 2000 hour rating in an accident situation. The full load, continuous operation rating for each diesel generator is 4700 kW, while the calculated accident loading in Modes 1 through 4 is 4000 kW. The largest anticipated load (including loads which are required to mitigate the consequences of a design basis accident or facilitate plant operation and maintenance) in Modes 5 and 6 is calculated to be less than 80% of the full rated capacity. No 2000 hour loading has been specified by the diesel generator manufacturer and, as a result the full loading rating of 4700 kW is conservatively established as the 2000 hour rating. Diesel frequency droop restrictions are established due to HPSI flow rate considerations.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery thermal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

AC SOURCES, DC SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (continued)

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.15 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufactuere's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Attachment D

ELECTRICAL POWER SYSTEMS

AC SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. Day fuel tanks containing a minimum volume of 325 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 37,600 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum AC electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and movement of irradiated fuel, or operation of the fuel handling machine with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required circuit between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at

SURVEILLANCE REQUIREMENTS (continued)

least once per 7 days by verifying correct breaker alignment and indicated power availability.

a. If the above required offsite source is supplied through the Unit 2 4160 volt Emergency Bus #2A04, the following buses are required:

   480 volt Emergency Bus #2B04
   125 volt Emergency Bus #2D1

b. If the above required offsite source is supplied through the Unit 2 4160 volt Emergency Bus #2A06, the following buses are required:

   480 volt Emergency Bus #2B06
   125 volt Emergency Bus #2D2

4.8.1.2.2 The above required diesel generator shall be demonstrated OPERABLE by performing the Surveillance Requirements of 4.8.1.1.2 (except 4.8.1.1.2 a.5, d.5, d.6, d.8, d.10 and d.12) and 4.8.1.1.3.

AC SOURCES, DC SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (continued)

as well as operation of loss of voltage logic, is the same as for the primary connection using the reserve auxiliary transformer, with the exception of no transfer to the companion unit.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Reg. Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975. However, ASTM-D270-1965 was reverified in 1975 rather than re-issued. The reverified 1965 standard is therefore the approproate standard to be used.

Additionally, Regulatory Guide 1.9 allows loading of the diesel generator to its 2000 hour rating in an accident situation. The full load, continuous operation rating for each diesel generator is 4700 kW, while the calculated accident loading in Modes 1 through 4 is 4000 kW. The largest anticipated load (including loads which are required to mitigate the consequences of a design basis accident or facilitate plant operation and maintenance) in Modes 5 and 6 is calculated to be less than 80% of the full rated capacity. No 2000 hour loading has been specified by the diesel generator manufacturer and, as a result the full loading rating of 4700 kW is conservatively established as the 2000 hour rating. Diesel frequency droop restrictions are established due to HPSI flow rate considerations.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery thermal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

AC SOURCES, DC SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (continued)

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.15 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufactuere's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

DESCRIPTION OF PROPOSED CHANGES
NPF-10-179 AND NPF-15-179
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3.9.10 to lower the required Reactor Vessel water level to couple and uncouple CEA's or to verify the coupling or uncoupling.

Existing Specifications

Unit 2: See Attachment "A"

Unit 3: See Attachment "B"

Proposed Specifications

Units 2 and 3: See Attachment "C"

Description

The amendment would change Technical Specification 3/4.9.10 (Refueling) Water Level - Reactor Vessel. Specification 3/4.9.10 requires that a minimum water level of 23 feet be maintained above the reactor vessel flange during movements of Control Element Assembilies (CEA's) or fuel assemblies in the reactor vessel. During refueling, the Control Element Drive Motor (CEDM) drive shaft extensions are uncoupled and recoupled to the CEA's. Coupling or uncoupling of the CEDM drive shaft extensions involve small movements of the CEA's as does the verification of coupling/uncoupling. The staff has interpreted that these small movements of the CEA's are within the applicability of Specification 3/4.9.10. Therefore under the current Technical Specification the water level of 23 feet must be maintained during these operations. The design of the tools used to couple and uncouple the CEA's from the CEDM drive shaft extensions require that the work platform be positioned less than 23 feet above the reactor vessel flange. Verification of CEA coupling/uncoupling is most efficiently accomplished when the CEA's are coupled/uncoupled.

The proposed change adds a note to the applicability for Specification 3/4.9.10 which allows the water level to be lowered to 23 feet above the fuel assemblies during CEA coupling and uncoupling and verification of coupling/uncoupling.

Safety Analysis

The proposed changes discussed shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

Operation during refueling will be as described in the FSAR section 9.1.4.2.3.3 which discusses the requirement to lower the work platform on the upper guide structure lift rig in order to disconnect the CEDM drive shaft extensions from their CEA's. Weighing of the CEA's to verify whether they are coupled or uncoupled is described in FSAR section 15.7.3.4.1. Thus, operation of the facility with this proposed change will be in accordance with the safety analysis and there will be no increase in the probability of an accident previously evaluated.

The basis for requiring 23 feet of water above the reactor vessel flange is to ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released by an irradiated fuel assembly striking the reactor vessel flange and rupturing. During coupling and uncoupling the CEA's or verifying the coupling or uncoupling, the fuel will be seated in the reactor pressure vessel. With the fuel seated in the pressure vessel, no fuel damage could occur above the top of the fuel. Thus requiring 23 feet of water above the top of the fuel will ensure sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from any conceivable accident and this proposed change will not increase the consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Operation of the facility will be as described in FSAR sections 9.1.4.2.3.3 and 15.7.3.4.1. Thus, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   Operation of the facility will be in accordance with the assumption made in the safety analysis and the bases of the Technical Specification that 23 feet of water will be available over any fuel damaged in a fuel handling accident. Thus, operation of the facility in accordance with the proposed change will not involve a reduction in a margin of safety.

The Commission has provided guidance for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously-analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

In this case, the acceptance criteria relating to refueling water level are delineated in the Bases Section of NUREG-0212, Revision 2, Standard Technical Specifications (STS) for Combustion Engineering Pressurized Water Reactors. Specifically, Bases Sections B 3/4.9.10, which requires that sufficient water depth (23 feet) is available to remove 99% of the assumed 10% iodine gap acticity which would be released by an irradiated fuel assembly striking the reactor vessel flange and rupturing. With fuel seated in the reactor vessel, as is the case with the proposed change, no fuel damage could occur above the top of the fuel. The proposed change's requirement to maintain 23 feet of water above the top of the fuel will ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from a fuel assembly damaged by any conceivable accident. Therefore, the proposed change meets the acceptance criteria delineated in the Bases of the STS and is similar to example (vi).

Safety and Significant Hazards Determination

Based on the above Safety Analysis it is concluded that; (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT A

(Existing Specification)

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

ATTACHMENT B

(Existing Specification)

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

NOV 15 1982

ATTACHMENT C

(Proposed Specifications)

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet* of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

* Water level may be lowered to at least 23 feet above the top of the fuel for coupling and uncoupling of CEA extension shafts or for verifying the coupling or uncoupling.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-180 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.9.12, Fuel Handling Building Post Accident Cleanup Filter System.

<u>Existing Specifications</u>:

Unit 2: See Attachment "A"
Unit 3: See Attachment "C"

<u>Proposed Specifications</u>:

Unit 2: See Attachment "B"
Unit 3: See Attachment "D"

<u>Description</u>:

The proposed change would revise Technical Specification (TS) 3/4.9.12, Fuel Handling Building Post Accident Cleanup Filter System (FHBPACFS). Specification 3/4.9.12 requires that Fuel Handling Building Post Accident Cleanup Filter System be operable when irradiated fuel is in the storage pool and defines a number of functional tests which periodically must be conducted to assure such operability. The FHBPACFS includes electrical heaters to maintain the relative humidity at the inlet to the charcoal filters at or below 70% to preserve charcoal adsorber efficiency. Specification 4.9.12.d.3 requires verification that the heater dissipation is within ±5% of the specified rating. The heater ratings contained in the specification are at the nominal operating voltage. However, when the plant is on line in normal operation, the bus voltages are higher than nominal. Additionally, Specification 4.8.1.1 (Diesel Generator surveillance requirements) would permit a ± 10% bus voltage variation during diesel generator operation. Because the power dissipated by a heater varies with the square of the voltage, small deviations from the nominal voltage (e.g., ± 2.5%) will result in heater dissipations outside of the T.S. allowable range, thereby rendering the system inoperable.

The proposed change revises Specification 4.9.12.d.3 to allow correction of measured heater dissipation to nominal voltage. In addition, the proposed change corrects a typographical error in the specified dissipation for heater E-464. Heater E-464 is actually rated at 28.7 kw versus the 28.4 kw listed currently.

<u>Safety Analysis</u>

The proposed changes discussed shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change affects the manner in which the results of heater dissipation surveillance are evaluated. The purpose of this surveillance is to verify that the heater dissipation is sufficient to maintain the relative humidity of air entering the charcoal adsorbers at or below 70%. The heater dissipation is specified at a nominal voltage. Verification of heater dissipation at this voltage provides assurance that the heater dissipation for the entire range of allowable voltages (nominal ± 10%) is adequate to ensure that the relative humidity requirement is met. The surveillance in effect measures the heater internal resistance which is essentially a constant and the characteristic of the heater which determines its heat dissipation at any given voltage. Therefore, measurement of heater dissipation at voltages other than nominal voltage is a valid measure of the heater internal resistance. Correction of the measured heater dissipation to correspond to the equivalent nominal voltage dissipation for comparison with the T.S. nominal voltage acceptance criteria, as allowed by the proposed change, will provide assurance that heater dissipation over the entire allowable voltage range is adequate to ensure that the relative humidity requirement is met. Therefore, the charcoal adsorber efficiency and results of the fuel handling accident analysis are unaffected by the proposed change.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not affect the configuration of the facility or the manner in which it is operated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change does not reduce the efficiency of the charcoal adsorber. Therefore, no margin of safety is reduced.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously-analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP). Example (i) relates to a change which is purely administrative: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

SRP Section 9.4.2, Spent Fuel Pool Area Ventilation System (SFPAVS) references Regulatory Guide 1.52 which recommends that heaters be installed in the SFPAVS upstream of the charcoal adsorbers of sufficient capacity to maintain the relative humidity below 70%, thereby preserving the efficiency of the charcoal adsorber. The proposed change affects the manner in which the results of the heater dissipation surveillance tests are evaluated to accommodate the allowed variations for the nominal bus voltage which may exist at the time the surveillance is conducted. The proposed change does not reduce the heater dissipation requirements and preserves the 70% relative humidity acceptance criteria. Therefore, the proposed change satisfies the SRP acceptance criteria and is similar to example (vi).

Additionally, the proposed change increases the required dissipation for heater E-464 from 28.4 kw to 28.7 kw. This corrects a typographical error. Therefore, this proposed change is similar to example (i).

PWS:1882F

Attachment "A"

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm ± 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.4 ± 1.5 kw for E464, 32.3 ± 1.7 kw for E465, and 3.8 ± 0.2 kw for E652 when tested in accordance with ANSI N510-1975.

Attachment "B"

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm ± 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.~~4~~ 7 ± 1.5 kw for E464, 32.3 ± 1.7 kw for E465, and 3.8 ± 0.2 kw for E652 when tested in accordance with ANSI N510-1975 with the measured heater dissipation corrected to correspond to nominal voltage.

Attachment "C"

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm ± 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.4 ± 1.5 kw for E464, 32.3 ± 1.7 kw for E465, and 3.8 ± 0.2 kw for E652 when tested in accordance with ANSI N510-1975.

NOV 15 1982

Attachment "D"

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm ± 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.~~4~~7 ± 1.5 kw for E464, 32.3 ± 1.7 kw for E465, and 3.8 ± 0.2 kw for E652 when tested in accordance with ANSI N510-1975 with the measured heater dissipation corrected to correspond to nominal voltage.

NOV 15 1982