



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
DEC 31 1985

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated April 23, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

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DEC 31 1985

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 92, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change in Technical Specifications is to become effective within 45 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications the licensee shall adhere to the Technical Specifications for the systems, components, or operation existing at the time. The period of time during changeover of systems, components or operation shall be minimized or compensated for by suitable temporary alternatives.
4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 31, 1985

rw
PWR#4:DPWR-A
DWigginton:kab
12/11/85

MD
PWR#4/DPWR-A
MDuncan
12/11/85

OELD
SEI
12/23/85

BJ
PWR#4/DPWR-A
BJYoungblood
12/19/85

1945
The following information was obtained from the records of the
Department of the Interior, Bureau of Land Management, on
the subject of the land described in the foregoing
instrument.

The land described in the foregoing instrument is
situated in the County of ... State of ... and is
owned by ... The land is being
conveyed to the ... for the purpose of
... The land is being conveyed to the ... for the purpose of
...

The land described in the foregoing instrument is
situated in the County of ... State of ... and is
owned by ... The land is being
conveyed to the ... for the purpose of
... The land is being conveyed to the ... for the purpose of
...

WITNESSED my hand and the seal of the
Department of the Interior, Bureau of Land Management,
this ... day of ... 1945.

Very truly yours,
[Signature]



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 3 1 1985

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated April 23, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

DEC 31 1985

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 77, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

- 3. The change in Technical Specifications is to become effective within 45 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications the licensee shall adhere to the Technical Specifications for the systems, components, or operation existing at the time. The period of time during changeover of systems, components or operation shall be minimized or compensated for by suitable temporary alternatives.
- 4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 31, 1985

DW
PWR#4:DPWR-A
DWigginton:kab
12/ 1 /85

MD
PWR#4/DPWR-A
MDuncan
12/ 1 /85

OELD
sc
12/ 23 /85

BJ
PWR#4/DPWR-A
BJYoungblood
12/ 31 /85

OFFICE OF THE SECRETARY OF DEFENSE

MEMORANDUM FOR THE SECRETARY OF DEFENSE
SUBJECT: [Illegible]

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ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 92 FACILITY OPERATING LICENSE NO. DPR-58

AMENDMENT NO. 77 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NOS. 50-315 AND 50-316

Revise Appendix A as follows:

Remove Pages

Unit 1

3/4 3-21a
3/4 7-5
3/4 7-6
B 3/4 7-1*
B 3/4 7-2
B 3/4 7-3
B 3/4 7-4*

Unit 2

3/4 3-20a
3/4 7-5
3/4 7-6
B 3/4 7-1*
B 3/4 7-2
B 3/4 7-3
B 3/4 7-4*

Insert Pages

Unit 1

3/4 3-21a
3/4 7-5
3/4 7-6
B 3/4 7-1*
B 3/4 7-2
B 3/4 7-3
B 3/4 7-4*

Unit 2

3/4 3-20a
3/4 7-5
3/4 7-6
B 3/4 7-1*
B 3/4 7-2
B 3/4 7-3
B 3/4 7-4*

*Included for convenience only

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. 4 kv Bus Loss of Voltage	2/Bus	2/Bus	2/Bus	1, 2, 3	14*
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2	18*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kv Bus Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*

D. C. COOK - UNIT 1

3/4 3-21a

Amendment No. 92

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

- a. Two feedwater pumps, each capable of being powered from separate emergency busses, and
- b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

- a. At least once per 31 days by:
 1. Verifying that each motor driven pump develops an equivalent discharge pressure of ≥ 1375 psig at 60°F on recirculation flow.
 2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of ≥ 1285 psig at 60°F and at a flow of ≥ 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.
 4. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER. This requirement is not applicable for those portions of the Auxiliary Feedwater System being used intermittently to maintain steam generator level.
- b. At least once per 18 months during shutdown by:
1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.
 2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is approximately 121 percent of the total secondary steam flow of 14,120,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line = 1, 2 or 3.

PLANT SYSTEMS

BASES

- U = maximum number of inoperable safety valves per operating steam line = 1, 2 or 3.
- (109) = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation
- (76) = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 3 loop operation.
- X = Total relieving capacity of all safety valves per steam line = 4,288,450 lbs/hour.
- Y = Maximum relieving capacity of any one safety valve = 857,690 lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1065 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1065 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

The acceptance discharge pressures for the auxiliary feedwater pumps are based on a fluid temperature of 60°F. Water density corrections are permitted to allow comparison of test results which vary depending on ambient conditions.

In addition to its safety design function, the AFW system is used to maintain steam generator level during startup (including low power operation). During this time, the system design allows for automatic initiation of the auxiliary feedwater pumps and their related automatic valves in the flow path.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

PLANT SYSTEMS

BASES

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values taken at +10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

The OPERABILITY of the essential service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. 4 kv Bus Loss of Voltage	2/Bus	2/Bus	2/Bus	1, 2, 3	14*
c. Safety Injection	2	1	2	1, 2, 3	18*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2	18*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	19*
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 Kv Bus Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

- 3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:
- a. Two feedwater pumps, each capable of being powered from separate emergency busses, and
 - b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

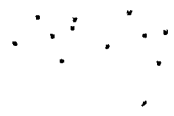
SURVEILLANCE REQUIREMENTS

- 4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:
- a. At least once per 31 days by:
 1. Verifying that each motor driven pump develops an equivalent discharge pressure of > 1375 psig at 60°F on recirculation flow.
 2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of > 1285 psig at 60°F and at a flow of > 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.
 4. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER. This requirement is not applicable for those portions of the Auxiliary Feedwater System being used intermittently to maintain steam generator level.
- b. At least once per 18 months during shutdown by:
1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.
 2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate engineered safety features actuation test signal required by Specification 3/4.3.2.



3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 17,153,800 lbs/hr which is 117 percent of the total secondary steam flow of 14,674,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

PLANT SYSTEMS

BASES

- U = maximum number of inoperable safety valves per operating steam line
- 109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation
- 76 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 3 loop operation.
- X = Total relieving capacity of all safety valves per steam line in lbs/hour = 4,288,450.
- Y = Maximum relieving capacity of any one safety valve in lbs/hour = 857,690.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1065 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1065 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

The acceptance discharge pressures for the auxiliary feedwater pumps are based on a fluid temperature of 60°F. Water density corrections are permitted to allow comparison of test results which vary depending on ambient conditions.

In addition to its safety design function, the AFW system is used to maintain steam generator level during startup (including low power operation). During this time, the system design allows for automatic initiation of the auxiliary feedwater pumps and their related automatic valves in the flow path.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

PLANT SYSTEMS

BASES

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 STEAM GENERATOR STOP VALVES

The OPERABILITY of the steam generator stop valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the steam generator stop valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

PLANT SYSTEMS

BASES

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values taken at +10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

The OPERABILITY of the essential service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room EMERGENCY ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.