3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 ~~All power~~ BOTH POWER-OPERATED relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or ~~more~~ BOTH PORV(s) inoperable BECAUSE OF EXCESSIVE SEAT LEAKAGE, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) WITH POWER MAINTAINED TO ~~and remove power from~~ the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

INSERT 1a

~~b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

REPLACE WITH INSERT 1b

f. ~~c.~~ The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by: ~~performance of a CHANNEL CALIBRATION and operating the valve through one cycle of full travel.~~

4.4.5.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed ~~with the power removed~~ in order to meet the ~~ACTION requirements of a. above.~~ REQUIREMENTS OF ACTION b. OR c. IN SPECIFICATION 3.4.5.

a. OPERATING THE PORV THROUGH ONE COMPLETE CYCLE OF FULL TRAVEL DURING MODE 3 OR 4, AND

b. OPERATING THE PORV THROUGH ONE COMPLETE CYCLE OF FULL TRAVEL USING THE BACKUP PORV CONTROL SYSTEM, AND

c. PERFORMING A CHANNEL CALIBRATION OF THE ACTUATION INSTRUMENTATION.

INSERT 1a

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1b

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the inoperable block valve 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.

e. With both block valves inoperable, within 1 hour restore both block valves to OPERABLE status or place their associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour and restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant Pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES (PORV's)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORV's minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. ~~Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.~~

INSERT 2

INSERT 2

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).

D. Manual control of a block valve to isolate a stuck-open PORV.

The Surveillance Requirements found in Specification 4.4.5.1 for the PORVs and Specification 4.4.5.2 for the block valves improve reliability and provide the assurance that the PORVs and block valves can perform their functions. The PORVs are stroke tested during MODES 3 or 4 with the associated block valves closed in order to eliminate uncertainty introduced by testing the PORVs at lesser system temperatures than expected during actual operating conditions. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status. Surveillance requirement 4.4.5.1.b includes testing which demonstrates the functionality of the backup PORV control system.

Unit 1

Typed Pages

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the inoperable block valve 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.

e. With both block valves inoperable, within 1 hour restore both block valves to OPERABLE status or place their associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour and restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODE 3 or 4, and

b. Operating the PORV through one complete cycle of full travel using the backup PORV control system, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.5.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b or c in Specification 3.4.5.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant Pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES (PORVs)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).

D. Manual control of a block valve to isolate a stuck-open PORV.

The Surveillance Requirements found in Specification 4.4.5.1 for the PORVs and Specification 4.4.5.2 for the block valves improve reliability and provide the assurance that the PORVs and block valves can perform their functions. The PORVs are stroke tested during MODES 3 or 4 with the associated block valves closed in order to eliminate uncertainty introduced by testing the PORVs at lesser system temperatures than expected during actual operating conditions. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status. Surveillance requirement 4.4.5.1.b includes testing which demonstrates the functionality of the backup PORV control system.

Farley Unit 2
Proposed Changed
Technical Specification Pages

Remove Page	Insert Page
3/4 4-8	3/4 4-8
---	3/4 4-8a
B 3/4 4-2	B 3/4 4-2
---	B 3/4 4-2a

Unit 2

Marked Pages

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 ~~All power~~ BOTH POWER-OPERATED relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or ~~more~~ BOTH PORV(s) inoperable, BECAUSE OF EXCESSIVE SEAT LEAKAGE within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) WITH POWER ~~and remove power from~~ MAINTAINED TO the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

INSERT 1a

~~b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

REPLACE WITH INSERT 1b

f. ~~e.~~ The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by: ~~performance of a CHANNEL CALIBRATION and operating the valve through one cycle of full travel.~~

4.4.5.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed ~~with the power removed~~ in order to meet the ~~ACTION requirements of a. above.~~ REQUIREMENTS OF ACTION b. OR c. IN SPECIFICATION 3.4.5.

a. OPERATING THE PORV THROUGH ONE COMPLETE CYCLE OF FULL TRAVEL DURING MODE 3 OR 4, AND

b. OPERATING THE PORV THROUGH ONE COMPLETE CYCLE OF FULL TRAVEL USING THE BACKUP PORV CONTROL SYSTEM, AND

c. PERFORMING A CHANNEL CALIBRATION OF THE ACTUATION INSTRUMENTATION.

INSERT 1a

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 1b

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the inoperable block valve 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.

e. With both block valves inoperable, within 1 hour restore both block valves to OPERABLE status or place their associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour and restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant Pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES (PORV's)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORV's minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. ~~Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.~~ INSERT 2

INSERT 2

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).

D. Manual control of a block valve to isolate a stuck-open PORV.

The Surveillance Requirements found in Specification 4.4.5.1 for the PORVs and Specification 4.4.5.2 for the block valves improve reliability and provide the assurance that the PORVs and block valves can perform their functions. The PORVs are stroke tested during MODES 3 or 4 with the associated block valves closed in order to eliminate uncertainty introduced by testing the PORVs at lesser system temperatures than expected during actual operating conditions. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status. Surveillance requirement 4.4.5.1 b includes testing which demonstrates the functionality of the backup PORV control system.

Unit 2

Typed Pages

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV in manual control. Restore the inoperable block valve 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.

e. With both block valves inoperable, within 1 hour restore both block valves to OPERABLE status or place their associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour and restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Operating the PORV through one complete cycle of full travel using the backup PORV control system, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.5.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b or c in Specification 3.4.5.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant Pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES (PORVs)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. This function has been classified as safety related for more recent plant designs.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).

D. Manual control of a block valve to isolate a stuck-open PORV.

The Surveillance Requirements found in Specification 4.4.5.1 for the PORVs and Specification 4.4.5.2 for the block valves improve reliability and provide the assurance that the PORVs and block valves can perform their functions. The PORVs are stroke tested during MODES 3 or 4 with the associated block valves closed in order to eliminate uncertainty introduced by testing the PORVs at lesser system temperatures than expected during actual operating conditions. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status. Surveillance requirement 4.4.5.1.b includes testing which demonstrates the functionality of the backup PORV control system.

Attachment 2

Safety Analysis

For Proposed Technical Specification Changes

Associated With Power-Operated Relief Valve And

Block Valve Reliability

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Changes Associated With
Generic Letter 90-06
Power-Operated Relief Valve and Block Valve Reliability

Safety Analysis

Proposed Change

Revise Farley Limiting Condition for Operation 3.4.5, Surveillance Requirements 4.4.5.1 and 4.4.5.2, and Bases 3/4.4.5 to incorporate the following changes:

1. Revised Limiting Condition for Operation (LCO) Action Statement "a." for Technical Specification 3.4.5 to specify that power be maintained to a block valve which is closed due to its associated PORV being inoperable due to excessive seat leakage. Additionally, revise the shutdown requirement for Action Statement "a." to require that the unit be placed in Hot Shutdown within the following 6 hours after reaching Hot Standby.

2. Added LCO Action Statement "b." for Technical Specification 3.4.5 to specify that power be removed from a block valve that is closed due to its associated PORV being inoperable for reasons other than excessive seat leakage. The Action Statement is applicable when only one PORV is inoperable and requires that the PORV be restored to operable status within 72 hours or be in Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours.

3. Added LCO Action Statement "c." for Technical Specification 3.4.5 to specify that in the event that both PORVs are inoperable for reasons other than excessive seat leakage, restore at least one PORV to operable status or close its associated block valve and remove power from the block valve and be in Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours.

4. Added LCO Action Statement "d." for Technical Specification 3.4.5 to specify that in the event that one block valve is inoperable, restore the block valve to operable status within 1 hour or place the associated PORV in manual control. The Action Statement requires that the inoperable block valve be restored 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.

5. Added LCO Action Statement "e." for Technical Specification 3.4.5 to specify that in the event that both block valves are inoperable, restore both block valves to operable status or place their associated PORV in manual control within 1 hour. Restore at least one block valve to operable within the next hour and restore the remaining inoperable block valve to operable status within 72 hours; otherwise, be in at least Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours.

6. Revised Surveillance Requirement 4.4.5.1 to reflect the requirement of Generic Letter 90-06 to operate the PORVs through one full cycle of travel at least once per 18 months while in Modes 3 or 4. In addition, added Surveillance Requirement 4.4.5.1.b requiring that the backup PORV control system be stroke tested at least once per 18 months.

7. Revised Surveillance Requirement 4.4.5.2 in order to prevent surveillance testing of the block valves when they are closed as a result of Action Statement b. or c. of Specification 3.4.5.

8. Revised the associated technical specification Bases to reflect the proposed changes and to better define the basis for operability of the PORVs and block valves. Clarified that the PORVs are stroke tested during MODES 3 or 4 with the associated block valves closed.

Basis and Justification

The proposed changes to the Farley Technical Specifications are consistent with the guidance provided by Generic Letter 90-06. The proposed changes to Specification 3.4.5 and the associated Bases increase the probability that the PORVs would be available in the event they were called upon to relieve RCS pressure. The proposed changes do not eliminate any function previously required by the PORVs, do not increase the probability of inadvertent opening of the PORVs, and do not create any new challenges to the RCS pressure boundary.

Attachment 3

Significant Hazards Evaluation

Pursuant to 10 CFR 50.92

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Changes Associated With
Generic Letter 90-06
Power-Operated Relief Valve and Block Valve Reliability

10 CFR 50.92 Evaluation

Proposed Changes

The proposed changes to the Farley Unit 1 and Unit 2 Technical Specifications are required in order to improve the reliability of the PORVs and block valves to ensure that the PORVs are available when called upon to perform their function. These changes involve the revision of Specification 3.4.5 and the associated Bases and are consistent with the recommendations of Generic Letter 90-06.

Background

The Nuclear Regulatory Commission (NRC) issued Generic Letter 90-06, Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressurization Pr[illegible] for Light-Water Reactors," June 25, 1990. The Generic Letter provided gu[illegible]e to licensees regarding the allowable outage time for inoperable PORVs and block valves in order to increase the reliability and availability of the PORVs and block valves. The proposed changes are consistent with the guidance provided by Generic Letter 90-06.

At Farley Nuclear Plant, the PORV's function is to automatically relieve RCS pressure below the pressurizer safety valve setpoint and to reduce RCS pressure upon demand by the operator. Automatic actuation of the PORVs is not assumed to mitigate the consequences of a design basis accident as described in Chapter 15 of the FSAR. The safety functions performed by the PORVs are: 1) inactive valves which form part of the RCS boundary, and 2) manual operation as required by emergency operating procedures. The PORVs are utilized to depressurize the RCS in the event of a steam generator tube rupture and during natural circulation; however, automatic actuation is not relied upon by the emergency operating procedures. Additionally, the PORVs are not utilized for low temperature overpressure protection; instead, the residual heat removal suction relief valves perform this function.

Analysis

The proposed change to the Technical Specifications will increase the availability of the PORVs and their associated block valves. The proposed change will allow continued operation with PORVs inoperable due to excessive seat leakage by closing the associated block valve with power maintained to the block valve. This change will allow operators to continue to have a readily available path for relieving the RCS pressure by opening the PORV block valves. In addition, the proposed change also revises the shutdown requirement from Cold Shutdown to Hot Shutdown to be consistent with the mode applicability requirements for PORV operability.

The above changes increase the probability that the PORVs would be available if needed to mitigate the consequences of a steam generator tube rupture or for RCS cooldown. No function previously required of the PORVs has been deleted nor has the probability of the inadvertent opening of the PORVs been increased. Additionally, these changes are consistent with the guidance provided by Generic Letter 90-06. Therefore, this change will not have an adverse affect on the health and safety of the public.

Southern Nuclear Operating Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes and has made the following determination:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Automatic actuation of the PORVs at Farley Nuclear Plant is not assumed to mitigate the consequences of a design basis accident as described in Chapter 15 of the FSAR. The safety functions performed by the PORVs are: 1) inactive valves which form part of the RCS boundary, and 2) manual operation as required by emergency operating procedures. The proposed changes will increase the reliability of the PORVs thus ensuring they are available to perform their function when required to do so. Therefore, it can be concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any physical changes to the PORVs or their setpoints. The proposed changes do not delete any function previously provided by the PORVs nor has the probability of inadvertent opening been increased. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Therefore, it can be concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety. The proposed changes increase the reliability of the PORVs thus ensuring their availability when called upon to perform their function and will not impact any safety analysis assumptions. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Conclusion

Based on the preceding analysis, Southern Nuclear Operating Company has determined that the proposed changes to the technical specifications will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, Southern Nuclear Operating Company has determined that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Attachment 4

Environmental Evaluation

For Proposed Technical Specification Changes

Associated With Power-Operated Relief Valves And

Block Valve Reliability

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Changes Associated With
Generic Letter 90-06
Power-Operated Relief Valve and Block Valve Reliability

Environmental Evaluation

Pursuant to 10 CFR 51.22(c)(9), the proposed license amendment can be categorically excluded from the requirement to perform an environmental assessment or an environmental impact statement based on the following evaluation:

Southern Nuclear Operating Company has determined that the proposed changes to the Farley Unit 1 and Unit 2 Technical Specifications, to increase the reliability of the PORVs and block valves consistent with the guidance provided by Generic Letter 90-06, do not affect the types or amounts of any radiological or non-radiological effluents that may be released offsite. No increase in individual or cumulative occupational radiation exposure will result from these changes. Additionally, these changes do not involve the use of any resources not previously considered in the Final Environmental statement related to the operation of Farley Nuclear Plant.

Based upon this evaluation, it can be concluded pursuant to 10 CFR 51.22(b) that it is not necessary to perform an environmental assessment or an environmental impact statement.

Attachment 5

Safety Analysis

For Technical Specification Changes Associated With

Low-Temperature Overpressure Protection

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Changes Associated With
Generic Letter 90-06
Low-Temperature Overpressure Protection

Safety Analysis

Proposed Change

Revise Farley Limiting Condition for Operation 3.4.10.3 and Basis 3/4.4.10 to incorporate the following changes:

1. Revise Limiting Condition for Operation (LCO) Action Statement "a." for Technical Specification 3.4.10.3 to reduce the allowed outage time for one RHR relief valve from the current 7 days to 24 hours unless: 1) the pressurizer level is reduced to 30 percent (cold calibrated), and 2) a dedicated operator is assigned to perform RCS pressure monitor and control functions.

2. Revise Basis for Technical Specification 3/4.4.10 to clarify the means of providing low-temperature overpressure protection for the limiting heat addition transient.

Basis and Justification

In response to NRC Generic Letter 90-06, SNC submitted a proposed change to the technical specifications to revise the allowable outage time (AOT) for having one inoperable low-temperature overpressure protection (LTOP) channel from the current 7-day requirement to a 24-hour requirement for water-solid conditions. For non-water-solid conditions, the AOT would remain at 7 days.

According to FSAR Section 5.2.2, RCS LTOP is provided, during startup and shutdown when the RCS is in a water-solid condition, by the two independent RHR suction relief valves. The Farley LTOP system and supporting analysis is based on the fact that there is sufficient capacity provided by one RHR relief valve to limit the effects of: 1) the worst case mass input transient (inadvertent start of charging pumps) and 2) the limiting heat addition transient (RCP start) provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F.

A pressurizer level of 30 percent (cold calibrated) was selected as the definition of water-solid conditions. This level was chosen to allow the operator sufficient time to respond to the overpressure event so that the limits of Appendix G are not violated. In order to evaluate the risk while operating under an LCO for one inoperable LTOP channel, the postulated failure of the other LTOP channel must be considered.

An analysis of the consequences of the inadvertent start of two charging pumps was performed assuming both LTOP channels are inoperable with no other RCS vents available and an initial pressurizer level of 30 percent. The analysis demonstrated that the limits of Appendix G would be exceeded within approximately 3.5 minutes.

Prior analysis of the limiting heat addition transient of an RCP with a temperature difference between the steam generators and the RCS primary side of less than 50°F concluded that an initial pressurizer level of 30 percent provides sufficient capacity for water expansion to prevent the limits of Appendix G from being exceeded.

Due to the short period of time in which an operator must respond to an inadvertent charging pump start, SNC proposes to dedicate an operator to the task of monitoring and controlling RCS pressure whenever an RHR suction relief valve is inoperable. This will provide greater assurance that the overpressure protection system is not challenged during the 7-day allowed outage time.

Based upon the above analysis, the proposed technical specification changes do not pose a safety concern.

Attachment 6

Proposed Changes

to the

Farley Units 1 and 2

Reactor Coolant System Overpressure Protection

Technical Specifications

Farley Unit 1
Proposed Changed
Technical Specification Pages

Remove Page	Insert Page
3/4 4-32	3/4 4-32
B 3/4 4-14	B 3/4 4-14
---	B 3/4 4-14a

Unit 1

Marked Page

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

1. A lift setting of less than or equal to 450 psig, and

2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.85 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 310°F, except when the reactor vessel head is removed.

ACTION: INSERT A

~~a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 7 days or depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.~~

b. With both RHR relief valves inoperable, within 8 hours either:

1. Restore at least one RHR relief valve to OPERABLE status, or

2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent.

c. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 1). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. As a result, such a fracture analysis was performed for Farley Unit 2. These Farley Unit 2 fracture analysis results are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. Based upon this fracture analysis, the 16 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of ~~two~~ either RHR relief valve~~s~~ or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of 3 charging pumps and their injection into a water solid RCS.

INSERT B

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10CFR50.44(c)(3)(iii).

INSERT A

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 24 hours or perform the following:

   1. Establish the following requirements:

      i. Reduce pressurizer level to less than or equal to 30 percent (cold calibrated), and

      ii. Assign a dedicated operator for RCS pressure monitoring and control, and

      iii. Restore the inoperable valve to OPERABLE status within 7 days, or;

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.

INSERT B

provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F,

Unit 1

Typed Page

LIMITING CONDITIONS FOR OPERATIONS

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

   1. A lift setting of less than or equal to 450 psig, and

   2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.85 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 310°F, except when the reactor vessel head is removed.

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 24 hours or perform the following:

   1. Establish the following requirements:

      i. Reduce pressurizer level to less than or equal to 30 percent (cold calibrated), and

      ii. Assign a dedicated operator for RCS pressure monitoring and control, and

      iii. Restore the inoperable valve to OPERABLE status within 7 days, or;

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.

b. With both RHR relief valves inoperable, within 8 hours either:

   1. Restore at least one RHR relief valve to OPERABLE status, or

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent.

c. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 1). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. As a result, such a fracture analysis was performed for Farley Unit 2. These Farley Unit 2 fracture analysis results are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. Based upon this fracture analysis, the 16 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of 3 charging pumps and their injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10 CFR 50.44(c)(3)(iii).

Farley Unit 2
Proposed Changed
Technical Specification Pages

Remove Page	Insert Page
3/4 4-32	3/4 4-32
B 3/4 4-14	B 3/4 4-14
---	B 3/4 4-14a

Unit 2

Marked Page

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

1. A lift setting of less than or equal to 450 psig, and

2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.85 square inches.

APPLICABLITY: When the temperature of one or more of the RCS cold legs is less than or equal to 310°F, except when the reactor vessel head is removed.

ACTION: INSERT A

~~a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 7 days or depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.~~

b. With both RHR relief valves inoperable, within 8 hours either:

1. Restore at least one RHR relief valve to OPERABLE status, or

2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent.

c. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 2). In addition, the new 10CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. Based upon such a fracture analysis for Farley Unit 2, the 14 EFPY heatup and cooldown curves are impacted by the new 10CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductible failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of ~~two~~ either RHR relief valves or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of 3 charging pumps and their injection into a water solid RCS. INSERT B

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10CFR50.44(c)(3)(iii).

INSERT A

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 24 hours or perform the following:

1. Establish the following requirements:

i. Reduce pressurizer level to less than or equal to 30 percent (cold calibrated), and

ii. Assign a dedicated operator for RCS pressure monitoring and control, and

iii. Restore the inoperable valve to OPERABLE status within 7 days, or;

2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.

INSERT B

provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F,

Unit 2

Typed Page

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

1. A lift setting of less than or equal to 450 psig, and

2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.85 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 310°F, except when the reactor vessel head is removed.

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 24 hours or perform the following:

1. Establish the following requirements:

i. Reduce pressurizer level to less than or equal to 30 percent (cold calibrated), and

ii. Assign a dedicated operator for RCS pressure monitoring and control, and

iii. Restore the inoperable valve to OPERABLE status within 7 days, or;

2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.

b. With both RHR relief valves inoperable, within 8 hours either:

1. Restore at least one RHR relief valve to OPERABLE status, or

2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent.

c. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 2). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. Based upon such a fracture analysis for Farley Unit 2, the 14 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of 3 charging pumps and their injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10 CFR 50.44(c)(3)(iii).

Attachment 7

Environmental Evaluation

For Proposed Technical Specification Changes

Associated With Low Temperature Overpressure Protection

Joseph M. Farley Nuclear Plant Units 1 and 2
Technical Specification Changes Associated With
Gereric Letter 90-06
Low-Temperature Overpressure Protection

Environmental Evaluation

Pursuant to 10 CFR 51.22(c)(9), the proposed license amendment can be categorically excluded from the requirement to perform an environmental assessment or an nvironmental impact statement based on the following evaluation:

Southern Nuclear Operating Company has determined that the proposed changes to the Farley Unit 1 and Unit 2 Technical Specifications, to increase the reliability of the RCS low-temperature overpressure protection system (LTOP) consistent with the guidance provided by Generic Letter 90-06, do not affect the types or amounts of any radiological or non-radiological effluents that may be released offsite. No increase in individual or cumulative occupational radiation exposure will result from these changes. Additionally, these changes do not involve the use of any resources not previously considered in the Final Environmental statement related to the operation of Farley Nuclear Plant.

Based upon this evaluation, it can be concluded pursuant to 10 CFR 51.22(b) that it is not necessary to perform an environmental assessment or an environmental impact statement.