Entergy

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T. A. Sullivan
Vice President, Operations–JAF

May 11, 2001
JAFP-01-0116

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station O-P1-17
Washington, DC 20555-0001

SUBJECT: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Proposed One-Time-Only Change to the Technical Specifications Regarding RHRSW Allowable Out-of-Service Time (JPTS-01-001)

Dear Sir:

This application for an amendment to the James A. FitzPatrick Technical Specifications (TS) proposes a change to Specification 3.5.B.3 and associated Bases. Specifically, this change extends the Limiting Condition for Operation (LCO) allowable out of service time for the Residual Heat Removal Service Water (RHRSW) system from 7 days to 11 days with special conditions to allow for installation of a modification to the division "B" RHRSW strainer.

The applicability of this proposed change is limited to the one-time-only installation of this modification on the "B" RHRSW strainer. A similar request was previously reviewed and approved by the NRC Staff for the "A" RHRSW strainer via TS Amendment No. 259 (dated January 28, 2000, TAC NO. MA6667). Entergy Nuclear Operations, Inc. requests that the NRC approve this request no later than July 19, 2001 to support the planned modification activities.

The signed original of the Application for Amendment to the Operating License is enclosed for filing. Attachment I contains the proposed new TS pages and Attachment II is the Safety Evaluation for the proposed changes. A markup of the affected TS pages is included as Attachment III.

A copy of this application and the associated attachments are being provided to the designated New York State official in accordance with 10 CFR 50.91.

There are no new commitments made in this letter. If you have any questions, please contact Mr. George Tasick at 315-349-6572.

Very truly yours,

(for)

T. A. Sullivan

cc: next page
Attachments as stated

A001

cc: Regional Administrator
U. S. Nuclear Regulatory Commission
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BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of)
Entergy Nuclear Operations, Inc.) Docket No. 50-333
James A. FitzPatrick Nuclear Power Plant)

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Entergy Nuclear Operations, Inc. requests an amendment to the Technical Specifications (TS) contained in Appendix A to Facility Operating License DPR-59 for the James A. FitzPatrick Nuclear Power Plant. This application is filed in accordance with Section 10 CFR 50.90 of the Nuclear Regulatory Commission's regulations.

This application for an amendment to the James A. FitzPatrick TS proposes a change to Specification 3.5.B.3 and associated Bases. Specifically, this change extends the Limiting Condition for Operation (LCO) for the Residual Heat Removal Service Water (RHRSW) system from 7 days to 11 days with special conditions to allow for installation of a modification to the division "B" RHRSW strainer.

The applicability of this proposed change is limited to the one-time-only installation of this modification on the "B" RHRSW strainer. A similar request was previously reviewed and approved by the NRC Staff for the "A" RHRSW strainer via TS Amendment No. 259 (dated January 28, 2000, TAC NO. MA6667).

The signed original of the Application for Amendment to the Operating License is enclosed for filing. Attachment I contains the proposed new TS pages and Attachment II is the Safety Evaluation for the proposed changes. A markup of the affected TS pages is included as Attachment III.

Entergy Nuclear Operations, Inc.

(for)

T. A. Sullivan
Vice President, Operations-JAF

STATE OF NEW YORK
COUNTY OF OSWEGO
Subscribed and sworn to before me this 11 day of May 2001.

Nancy B. Czerow
Notary Public

NANCY B. CZEROW
Notary Public, State of New York
Qualified in Oswego County #4884611
Commission Expires 1-26-03

Attachment I to JAFP-01-0116

REVISED TECHNICAL SPECIFICATION PAGES

ONE-TIME-ONLY CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING RHRSW ALLOWABLE OUT-OF-SERVICE TIME

(JPTS-01-001)

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

3.5 (cont'd)

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

3. Should one of the containment cooling subsystems become inoperable or should one RHRSW pump in each subsystem become inoperable, continued reactor operation is permissible for a period not to exceed 7 days.*

4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within 24 hr.

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $<212°F$ with an inoperable component(s) as specified in 3.5.B above.

4.5 (cont'd)

	Item	Frequency
e.	a verification that each valve (manual, power operated, or automatic) in the flowpath that is not locked, sealed or otherwise secured in position, is in the correct position.	Once per 31 Days
f.	an air test shall be performed on the containment spray headers and nozzles.	Once per 5 Years

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter. When one RHRSW pump in each subsystem becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.

During the installation of modification 00-125 to the "B" RHRSW strainer, continued reactor operation is permissible for a period not to exceed 11 days.

3.5 BASES (cont'd)

B. Containment Cooling Mode (of the RHR System)

The containment heat removal portion of the LPCI/containment spray mode is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability.

Each subsystem of the containment cooling mode (of the RHR System) consists of two RHR Pumps, two RHR service water pumps, one heat exchanger and a flowpath capable of recirculating water from the suppression pool through the heat exchanger and back to primary containment. Either subsystem is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability as any two of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a thirty-day repair period is adequate. Loss of one subsystem of the containment cooling mode leaves one remaining system to perform the containment cooling function. The operable system is verified to be operable each day when the above condition occurs. Based on the fact that when one containment cooling subsystem becomes inoperable only one system remains, a seven day repair period was specified.*

Low power physics testing and reactor operator training with inoperable components will be conducted only when the containment cooling mode of RHR is not required for the safety of the plant.

Calculations have been made to determine the effects of the design basis LOCA while conducting low power physics testing or operator training at or below 212°F. The results of these conservative calculations show that the suppression pool water temperature will not exceed 170°F. Therefore LPCI and Core Spray Systems will not be adversely affected by the postulated LOCA.

****During the installation of modification 00-125 to the "B" RHRSW strainer, the seven day repair period may be extended to eleven days. The Conditional Core Damage Probability with the plant in this configuration for eleven days has been determined to be below the threshold probability of 1 E-6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide.***

Attachment II to JAFP-01-0116

SAFETY EVALUATION

ONE-TIME-ONLY CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING RHRSW ALLOWABLE OUT-OF-SERVICE TIME

(JPTS-01-001)

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

I. DESCRIPTION

This proposed change to the James A. FitzPatrick Nuclear Power Plant (JAF) Technical Specifications (TS) extends the Limiting Condition for Operation (LCO) for the Residual Heat Removal (RHR) Service Water (RHRSW) system from 7 days to 11 days with special conditions to allow for installation of a modification to the division "B" RHRSW strainer.

The applicability of this proposed change is limited to the one-time-only installation of this modification on the "B" RHRSW strainer. A similar request was previously reviewed and approved by the NRC Staff for the "A" RHRSW strainer via TS Amendment No. 259 (Reference 5).

The Specific changes are as follows:

Technical Specification 3.5.B.3

Modify the following qualifying note to address the "B" RHRSW strainer:

> *"During the installation of modification 00-125 to the "B" RHRSW strainer, continued reactor operation is permissible for a period not to exceed 11 days."*

Bases for Technical Specification 3.5.B.

Modify the following qualifying note to address the "B" RHRSW strainer:

> *"During the installation of modification 00-125 to the "B" RHRSW strainer, the seven day repair period may be extended to eleven days. The Conditional Core Damage Probability with the plant in this configuration for eleven days has been determined to be below the threshold probability of 1 E-6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide."*

II. PURPOSE OF THE PROPOSED CHANGE

Each loop of the JAF RHRSW system has two RHRSW pumps, configured with a common header. The common header discharges into a duplex strainer, which then discharges to the system loads. Each duplex strainer is configured with a flow porting mechanism for directing RHRSW flow through either of the two strainer baskets to allow on-line cleaning of the other basket. The flow porting mechanism consists of two opposing crank operated piston/cylinder assemblies, which perform the flow porting function. Each piston/cylinder assembly is configured with a compression packing/stuffing box to minimize leakage between the piston ram and the strainer body.

The packing gland on the "B" RHRSW strainer is degrading due to corrosion. In order to correct this degrading condition, a permanent modification has been developed which will replace the degrading packing gland with that of a new design. This new packing gland design is a significant improvement over the existing design in terms of leak-tight performance and reliability.

The JAF staff has estimated that the RHRSW strainer modification installation will require approximately 5 days to complete. The current LCO allowable out of service time for a single division of the RHRSW system is 7 days. The projected modification installation schedule does not allow for unforeseen complications during modification installation, which could extend the RHRSW system outage beyond the 7 day LCO allowable out of service time in the current TS and would therefore result in a forced plant shutdown.

This proposed change to the JAF TS allows for unforeseen complications in the modification installation schedule described above. The applicability of this proposed change is limited to the one-time-only installation of the RHRSW strainer modification on the division "B" of RHRSW.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGE

This proposed change extends the LCO for the RHRSW system from 7 days to 11 days with special conditions to allow for installation of a modification to the division "B" RHRSW strainer. This modification corrects a degrading condition in the "B" RHRSW strainer by replacing the existing packing gland with a packing gland with an improved design, and therefore results in improved safety system reliability.

The RHRSW system is designed to provide cooling water to the RHR system heat exchangers required for normal reactor shutdown cooling and for safe shutdown following a design-basis accident or transient. The RHRSW system is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR system. The RHRSW system circulates service water through the tube side of the RHR heat exchangers, and supports long term cooling of the reactor or containment by exchanging heat with the reactor coolant or suppression pool water, and discharging this heat to the external heat sink. The RHRSW system consists of two 100 percent capacity, totally independent supply loops. Each of the independent loops is supplied from two RHRSW pumps. Each pair of pumps is powered from a separate emergency bus connected to the emergency diesel generators (EDGs). Only one of the two parallel loops is necessary for safe shutdown.

Probabilistic Risk Assessment

An evaluation using the JAF full-power IPE model (Reference 3) was performed which assessed the resultant core damage frequency (CDF), conditional core damage probability (CCDP), large early release frequency (LERF), and conditional large early release probability (CLERP). The evaluation is consistent with Regulatory Guide 1.174 (Reference 4) requirements. Additional information about the JAF specific PRA was provided to the NRC Staff to support the previous TS Amendment review in the New York Power Authority letter JAFP-99-0319, dated December 7, 1999. The general information provided about the JAF PRA in that letter is still applicable.

Although this letter requests an allowable out-of-service time of only 11 days, a 14-day period was used in the risk evaluation for additional conservatism. If concurrent TS allowed on-line maintenance for risk-significant SSCs is assumed, the increase in CDF over the base is 1.52 E-6 per year with a resultant CCDP of 5.83 E-8 for a 14-day period. If no risk-significant SSC maintenance is allowed, the increase over the base is 1.13 E-6 per year with a CCDP of 4.33 E-8 for a 14-day period. In either case, the CCDP falls below the EPRI PSA Applications Guide (Reference 2) threshold for risk-significance.

In the base IPE submittal, the large early release frequency (LERF) was estimated to be 6.62 E-7 per year and is dominated by Station Blackout (SBO) sequences. The contribution of loss of decay heat removal (TW) sequences to LERF is 6.97 E-8 per year which is 10.5% of the total frequency. For CDF, TW sequences contribute 2.72 E-7 per year, or approximately 12 percent to the total point-estimate CDF. A quantification of TW sequences leading to core damage was performed with an unavailable loop "B" of RHRSW. The resultant CDF contribution is 1.08 E-5 per year. The conditional probability for TW sequences is 0.22. Therefore, the LERF contribution for failure of TW is the product of the CDF contribution and the conditional probability, or 2.38 E-6 per year. This is an increase in TW LERF of 2.31 E-6 per year over the base. For a maximum planned 14-day allowable out of service time, the CLERP is estimated to be:

$$\frac{\text{14 days}}{\text{365 days/year}} \times \text{2.31 E-6 per year, or 8.85 E-8.}$$

This probability falls below the EPRI PSA Applications Guide threshold of E-7 for risk significance in a change in CLERP.

Risk Management

An assessment of dominant cut set contributors was performed. Due to the electrical configurations of RHR and RHRSW, initiated failures of Division I AC (71-10500) and DC (71BCB-2A) predominate. For post-accident events, failure to vent containment locally via EOP support procedure EP-6 (Post Accident Containment Venting and Gas Control) predominates. Therefore, to reduce the LERF contribution, no other maintenance activities will be scheduled during this period which can result in unavailability of Division I AC and DC. Operations department personnel will be briefed on performing procedure EP-6 actions locally at valves 27AOV-117 and 27AOV-118 (Torus Exhaust Isolation Valves), should the need arise. Plant risk will be managed during this proposed, extended LCO timeframe in accordance with JAF's existing Configuration Risk Management Program (CRMP) (Reference 1). This is an appropriate control because the CDF used to compute the CCDP due to this proposed change is consistent with the existing CRMP. As noted above, the sequences of importance during this LCO are SBO and loss of Division I AC and DC. As part of the CRMP, work activities during the LCO will be evaluated to ensure:

- planned activities that have a potential to result in a plant transient, Reactor Protection System (RPS) actuation, Primary Containment Isolation Control System (PCICS) trip, Emergency Core Cooling System (ECCS) actuation, or failure, are compatible with the planned LCO

- no planned degradation, through testing or maintenance, of any other safety function is scheduled or permitted

- no planned degradation of the electrical power distribution safety function is scheduled or permitted

Contingency Plan

An element of this program is the development of a contingency plan for this plant configuration. For the RHRSW modification installation, the contingency plan will provide for an alternate means of achieving RHRSW flow via a connection to the Fire Protection system header in the "A" loop. This will provide a means of achieving limited containment cooling in the event the "A" division of RHRSW were rendered inoperable for some reason during the "B" RHRSW strainer modification.

This connection is achieved by connecting a temporary hose to cross tie connections, which are permanently installed, between the "A" RHRSW loop and the Fire Protection system. Instructions for installing this crosstie exist in current plant procedures and the material required for crosstie installation is permanently pre-staged in an equipment cabinet in the RHRSW pump room. The "A" RHRSW loop pressure boundary will not be impacted by the work on the "B" loop, and thus the crosstie installation would allow the interface between the Fire Protection system and RHRSW without any other temporary connections or plugs.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the JAF plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

Involve an increase in the probability or consequences of an accident previously evaluated.

The CCDP due to this proposed change is calculated to be 4.33 E-8 (assuming no-risk significant SSC maintenance), which falls below the threshold probability of 1 E-6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide (Reference 2). The CLERP is calculated to be 8.85 E-8, which falls below the threshold probability of 1 E-7 for risk significance per Reference 2.

This proposed change does not increase the consequences of an accident previously evaluated because all relevant accidents (LOCA) would result in the transfer of decay heat to the suppression pool. For this scenario, the same compliment of equipment will be available to achieve and maintain cold shutdown as is required by the current TS LCO.

Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not physically alter the plant. As such, no new or different types of equipment will be installed. The new design for the RHRSW strainer packing gland will be evaluated under a separate 10 CFR 50.59 evaluation and is considered to be functionally equivalent for the purposes of this one-time-only proposed TS change.

The connection and use of a temporary hose for achieving limited containment heat removal in the event the "A" division of RHRSW is rendered inoperable for some reason is a contingency plan that is already addressed by current plant procedures.

Involve a significant reduction in a margin of safety.

The CCDP due to this proposed change is calculated to be 4.33 E-8 (assuming no-risk significant SSC maintenance). This value falls below the threshold probability of 1 E-6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide (Reference 2). The CLERP is calculated to be 8.85 E-8, which falls below the threshold probability of 1 E-7 for risk significance per Reference 2.

The consequences of a postulated accident occurring during the extended allowable out-of-service time are bounded by existing analyses, therefore, there is no significant reduction in a margin of safety.

V. IMPLEMENTATION OF THE PROPOSED CHANGE

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards consideration.

As described in Section IV of this evaluation, the proposed change involves no significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment that may affect the types or amounts of effluents that may be released offsite. Therefore, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not involve plant physical changes, or introduce any new mode of plant operation. Therefore, there is no significant increase in individual or cumulative occupational radiation exposure.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed changes meet the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to requiring a specific environmental assessment by the Commission.

VI. CONCLUSION

Removing one division of RHRSW from service with the plant at power is currently evaluated in support of the JAF CRMP. An evaluation using the JAF full-power IPE model was performed which assessed the resultant CDF, CCDP, LERF, and CLERP. As described above, the CCDP and change in CLERP fall below the EPRI PSA Applications Guide threshold for risk significance. Therefore, the allowable out-of-service time extension is not considered to be risk significant.

In addition, Modification 00-125 corrects a degrading condition in the "B" RHRSW strainer by replacing the existing packing gland with a packing gland with an improved design, and therefore results in improved safety system reliability.

The Plant Operating Review Committee (PORC) and Safety Review Committee (SRC) have reviewed this proposed change to the TS and have concluded that it does not involve a significant hazards consideration and will not endanger the health and safety of the public.

VII. REFERENCES

1. James A. FitzPatrick Nuclear Power Plant, Administrative Procedure AP-10.02, Rev. 13, "13-Week Rolling Schedule", December, 2000

2. EPRI TR-105396, PSA Applications Guide, August 1995

3. JAF-RPT-MULTI-02107, Rev. 1, "JAF Individual Plant Examination", April 1998

4. USNRC, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Regulatory Guide 1.174, July 1998

5. TS Amendment No. 259, dated January 28, 2000, "James A. FitzPatrick Nuclear Power Plant - Issuance of Amendment Re: One Time Residual Heat Removal Service Water Allowed Outage Time to Allow Implementation of a Modification to the "A" Residual Heat Removal Service Water (RHRSW) Strainer (TAC NO. MA6667)"

Attachment III to JAFP-01-0116

MARKED-UP TECHNICAL SPECIFICATION PAGES

ONE-TIME-ONLY CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING RHRSW ALLOWABLE OUT-OF-SERVICE TIME

(JPTS-01-001)

Entergy Nuclear Operations, Inc.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

3.5 (cont'd)

2. Should one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining components of the containment cooling mode subsystems are operable.

3. Should one of the containment cooling subsystems become inoperable or should one RHRSW pump in each subsystem become inoperable, continued reactor operation is permissible for a period not to exceed 7 days.*

4. If the requirements of 3.5.B.2 or 3.5.B.3 cannot be met, the reactor shall be placed in a cold condition within 24 hr.

5. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature <212°F with an inoperable component(s) as specified in 3.5.B above.

4.5 (cont'd)

	Item	Frequency
e.	a verification that each valve (manual, power operated, or automatic) in the flowpath that is not locked, sealed or otherwise secured in position, is in the correct position.	Once per 31 Days
f.	an air test shall be performed on the containment spray headers and nozzles.	Once per 5 Years

2. When it is determined that one RHRSW pump of the components required in 3.5.B.1 above is inoperable, the remaining components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3. When one containment cooling subsystem becomes inoperable, the redundant containment cooling subsystem shall be verified to be operable immediately and daily thereafter. When one RHRSW pump in each subsystem becomes inoperable, the remaining components of the containment cooling subsystems shall be verified to be operable immediately and daily thereafter.

**During the installation of modification 99-095 to the "A" RHRSW strainer, continued reactor operation is permissible for a period not to exceed 11 days.*

00-125

"B"

3.5 BASES (cont'd)

B. Containment Cooling Mode (of the RHR System)

The containment heat removal portion of the LPCI/containment spray mode is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability.

Each subsystem of the containment cooling mode (of the RHR System) consists of two RHR Pumps, two RHR service water pumps, one heat exchanger and a flowpath capable of recirculating water from the suppression pool through the heat exchanger and back to primary containment. Either subsystem is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability as any two of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a thirty-day repair period is adequate. Loss of one subsystem of the containment cooling mode leaves one remaining system to perform the containment cooling function. The operable system is verified to be operable each day when the above condition occurs. Based on the fact that when one containment cooling subsystem becomes inoperable only one system remains, a seven day repair period was specified.*

Low power physics testing and reactor operator training with inoperable components will be conducted only when the containment cooling mode of RHR is not required for the safety of the plant.

Calculations have been made to determine the effects of the design basis LOCA while conducting low power physics testing or operator training at or below 212°F. The results of these conservative calculations show that the suppression pool water temperature will not exceed 170°F. Therefore LPCI and Core Spray Systems will not be adversely affected by the postulated LOCA.

00-125 "B"

During the installation of modification 99-095 to the "A" RHRSW strainer, the seven day repair period may be extended to eleven days. The Conditional Core Damage Probability with the plant in this configuration for eleven days has been determined to be below the threshold probability of 1 E-6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide.