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

December 2, 1991

Docket No. 50-333

Posted
Amdt. 173 to DPR-59

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M79489)

The Commission has issued the enclosed Amendment No. 173 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 16, 1991, and supplemented by letter dated October 10, 1991.

The amendment deletes Table 3.7-1, "Primary Containment Isolation Valves," along with any associated notes and references to this table from the Technical Specifications. These changes were made in accordance with the guidance given in NRC Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991. Several administrative changes have also been incorporated in this amendment.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Brian C. McCabe

Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 173 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. DPR-59

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated January 16, 1991, and supplemented October 10, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 173, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 2, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 173

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
vi	vi
55	55
56	56
185	185
186	186
192	192
196	196
197	197
198	198
199	-
200	-
201	-
202	-
203	-
204	-
205	-
206	-
206a	-
206b	-
206c	-
206d	-
206e	-
206f	-
207	-
208	-
209	-

LIST OF TABLES (Cont'd)

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the Core Cooling Systems, Control Rod Block and Standby Gas Treatment Systems. The objectives of the specifications are to assure the effectiveness of the protective instrumentation when required, even during periods when portions of such systems are out of service for maintenance, and to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2-1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177 in. above the top of the active fuel closes all isolation valves except those in Group 1. Details of valve grouping are given in the JAF FSAR section 7.3. For valves which isolate at this level, this trip setting is adequate to prevent uncovering the core in the case of a break in the largest line.

The low-low reactor water level instrumentation is set to trip when reactor water level is 126.5 in. above the top of active fuel. This trip initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of active fuel. This trip activates the remainder of the ECCS subsystems, closes the main steam isolation valves, main steam line drain valves and reactor water sample line isolation valves, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and C isolation valves. For the breaks discussed above, this instrumentation will generally initiate ECCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. Details of the isolation valve closure group are given in the JAF FSAR section 7.3. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperature peak at approximately 1,000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section 14.6.5 FSAR.

3.7 (cont'd)

D. Primary Containment Isolation Valves

1. Whenever primary containment integrity is required per 3.7.A.2, containment isolation valves and all instrument line excess flow check valves shall be operable, except as specified in 3.7.D.2. The containment vent and purge valves shall be limited to opening angles less than or equal to that specified below:

Valve Number	Maximum Opening Angle
27AOV-111	40°
27AOV-112	40°
27AOV-113	40°
27AOV-114	50°
27AOV-115	50°
27AOV-116	50°
27AOV-117	50°
27AOV-118	50°

4.7 (cont'd)

c. Secondary containment capability to maintain a 1/4 in. of water vacuum under calm wind conditions with a filter train flow rate of not more than 6,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle, the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and for closure time.

b. At least once per operating cycle, the instrument line excess flow check valves shall be tested for proper operation.

c. At least once per quarter:

(1.) All normally open power-operated isolation valves (except for the main stream line and Reactor Building Closed Loop Cooling Water System (RBCLCWS) power-operated isolation valves) shall be fully closed and reopened.

3.7 (cont'd)

2. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve operable in each affected penetration that is open and:

   a. Restore the inoperable valve(s) to operable status within 4 hours; or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the closed position. Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control; or

   c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or a blind flange.

3. If Specifications 3.7.D.1 or 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold condition within 24 hrs.

4.7 (cont'd)

(2.) With the reactor at reduced power level, trip main steam isolation valves and verify closure time.

d. At least twice per week, the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

e. The RBCLCWS isolation valves shall be fully closed and reopened any time the reactor is in the cold condition exceeding 48 hours, if the valves have not been fully closed and reopened during the preceding 92 days.

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

3.7 BASES (cont'd)

of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the Pressure Suppression System. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The containment isolation valves on the containment vent and purge lines may be open for safety related reasons. Safety related reasons include, but are not limited to, the following: inerting or de-inerting primary containment; maintaining containment oxygen concentration; maintaining drywell and suppression pool atmospheric pressures; and maintaining the differential pressure between the drywell and suppression pool. These valves have been modified to limit the maximum angle of opening as shown in 3.7.D.1.

Nine remote manual isolation valves have been added to the Reactor Building Closed Loop Cooling Water System (RBCLCWS) in order to comply with 10 CFR 50 Appendix A GDC 57; These valves are air operated (with solenoid pilot valves), normally open, and are designed to fail "open"on loss of electrical power or "as is" upon loss of instrument air. Each AOV is provided with a Seismic Class I accumulator tank to allow operation of the valves upon loss of instrument air up to 2 full valve cycles. The fail-open design permits continued operation of the system to supply water to the recirculation pump-motor coolers and drywell coolers during normal operation and as necessary under accident conditions. If there is a postulated accident, and indications of leakage from RBCLCWS appear, the operator will selectively close the AOV's affected to provide containment isolation.

A list of containment isolation valves, including a brief description of each valve is included in the updated JAF FSAR section 7.3.

4.7 BASES (cont'd)

by in-place testing with DOP as testing medium.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, retention capacity in terms of milligrams of iodine per gram of charcoal will be demonstrated. This will be done by testing the charcoal once a year, unless filter efficiency seriously deteriorates. Since shelf lives greater than 5 yr. have been demonstrated, the test interval is reasonable.

D. Primary Containment Isolation Valves

The large pipes comprising a portion of the Reactor Coolant System, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for Emergency Core Cooling Systems operation or containment cooling). Valve closure times are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, isolation valve closure times are sufficient to prevent uncovering the core.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 sec.

For Reactor Coolant System temperatures less than 212°F, the containment could not become pressurized due to a loss-of-coolant accident. The 212°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

The primary containment isolation valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve

4.7 BASES (cont'd)

operability results in a more reliable system.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 in. restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

The RBCLCWS valves are excluded from the quarterly surveillance requirements because closure of these valves will eliminate the coolant flow to the drywell air and recirculation pump-motor coolers. Without cooling water, the drywell air and equipment temperature will increase and may cause damage to the equipment during normal plant operations. Therefore, testing of these valves would only be conducted in the cold condition.

A list of containment isolation valves, including a brief description of each valve is included in the updated JAF FSAR section 7.3.

PAGES 198 THROUGH 209

HAVE BEEN DELETED

Amendment No. ~~48~~, ~~91~~, ~~118~~, ~~150~~, 173

(Next page is 210)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 173 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

On May 6, 1991, the NRC staff issued Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications." This generic letter provides guidance for preparing a request for a license amendment to remove component lists from the technical specifications (TS). This guidance includes the incorporation of lists into plant procedures that are subject to the change control provisions in the Administrative Controls Section of the TS. The removal of component lists from the TS permits administrative control of changes to these lists without processing a license amendment, as is required to update TS component lists. Therefore, the change control provisions of the TS provide an adequate means to control changes to these component lists without including them in the TS.

By letter dated January 16, 1991, and supplemented by letter dated October 10, 1991, the Power Authority of the State of New York (PASNY or the licensee) submitted a proposed amendment requesting changes to the technical specifications for the James A. FitzPatrick Nuclear Power Plant. The proposed amendment would remove TS Table 3.7-1, "Primary Containment Isolation Valves," and delete any references to it from the technical specifications. The purpose of TS Table 3.7-1 is to maintain a current listing of those primary containment isolation valves which must close to ensure that primary containment integrity is maintained during a severe transient or accident. The licensee's proposal would relocate the list of containment isolation valves and the associated information in TS Table 3.7-1 to a plant procedure that is subject to the change control provisions in the Administrative Controls Section of the technical specifications. In addition, the list of the primary containment isolation valves has also been incorporated into the Final Safety Analysis Report (FSAR). These changes will assure that the table is adequately controlled without the administrative burden of an operating license amendment. Furthermore, the proposed revision will not remove the TS requirement to maintain primary containment integrity and the associated Limiting Conditions for Operation (LCOs) nor will it remove the requirement to perform surveillance testing periodically and after maintenance, repair, or replacement work to demonstrate operability of the containment isolation valve.

Specifically, the proposed change would: (1) delete the reference to Table 3.7-1 from the List of Tables on page vi; (2) replace the tables and notes on pages 198 through 209 with a note stating that the pages have been deleted; (3) delete an erroneous reference to a list of containment isolation valve closure times in Bases 4.7.D from page 196; (4) delete references to Table 3.7-1 from pages 185 and 186, (5) delete references to TS 3.7 from and correct administrative errors on Bases pages 55 and 56; and (6) incorporate references to FSAR Section 7.3 into Bases pages 55, 56, 192, and 197.

2.0 EVALUATION

The licensee has proposed the removal of Table 3.7-1, "Primary Containment Isolation Valves," and the deletion of any references to it from the technical specifications. Although all references to Table 3.7-1 have been removed, the LCO action statements (TSs 3.7.D.1 and 3.7.D.2) and the surveillance requirements (TSs 4.7.D.1.a and 4.7.D.2) for primary containment isolation valves remain unchanged. Furthermore, the requirements to ensure "Primary Containment Integrity" and "Operability," as defined in TS 1.0, are unaffected by this proposed amendment.

The information presently contained in Table 3.7-1 will be transferred and maintained in controlled documents (the FSAR and a plant procedure). The licensee will maintain the table of containment isolation valves in a plant procedure that is subject to the change control provisions in the Administrative Controls Section of the technical specifications. TS Section 6.8, "Procedures," includes provisions for the control of plant procedures. Section 6.8.(B) requires that procedures affecting nuclear safety be reviewed by the Plant Operating Review Committee and approved by the Resident Manager prior to implementation. Moreover, any future changes to the design or operating characteristics of primary containment isolation valves will be made subject to the provisions of 10 CFR 50.59, "Changes, tests, and experiments" and 10 CFR 50.71(e), "Maintenance of records, making of reports."

The Code of Federal Regulations at 10 CFR 50.59 provides that any proposed change will be reviewed by the licensee for its potential impact on safety. It also requires that the licensee complete a safety evaluation for each change to assure that the change does not involve an unresolved safety question. Furthermore, this section requires the licensee to annually submit to the NRC a report containing a description of such changes including a summary of the safety evaluation for each. The Code of Federal Regulations at 10 CFR 50.71 requires the licensee to revise the FSAR each year to assure that the information included in the FSAR contains the latest material developed. These requlations ensure that there is sufficient engineering evaluation and management oversight of changes made which affect the operation of primary containment isolation valves.

The licensee has proposed changes to the above TSs that are consistent with the guidance provided in Generic Letter 91-08. The licensee has confirmed that the component list removed from the TS identifies all components for which the TS requirements apply and is located in a controlled plant procedure. In addition, the list of primary containment isolation valves has been incorporated into Section 7.3 of the FSAR.

The proposed changes to TS Bases 4.7.D (page 196) delete an erroneous reference to a list of containment isolation valve closure times. The NRC staff has determined that these changes are administrative and will have no impact on the design or operability requirements of any primary containment isolation valve.

On the basis of its review of this matter, the NRC staff finds that the proposed changes to the TS for the James A. FitzPatrick plant are primarily administrative changes and do not alter the requirements set forth in the existing TS. Overall, these changes will allow the licensee to make corrections and updates to the list of components for which these TS requirements apply, under the provisions that control changes to plant procedures as specified in the Administrative Controls Section of the TS. Therefore, the staff finds that the proposed TS changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 60880 and 56 FR 55949). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
B. McCabe

Date: December 2, 1991

December 2, 1991

Docket No. 50-333

DISTRIBUTION:
See attached sheet

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

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A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original Signed By:

Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 173to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFC	:PDI-1:LA	:PDI-1:PM	:*OGC	:PDI-1:D	:
NAME	:CVogan CV	:BMcCabe BM	:	:RCapra RAC	:
DATE	:12/3/91	:12/2/91	:11/5/91	:12/2/91	:

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