ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING THE DELETION OF TABLE 3.7-1 "PRIMARY CONTAINMENT ISOLATION VALVES"

JPTS-89-027

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

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 times as specified in the JAF FSAR section 7.3.

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 listed in the JAF FSAR section 7.3 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 listed in the JAF FSAR section 7.3 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)

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

ATTACHMENT II

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING THE DELETION OF TABLE 3.7-1 "PRIMARY CONTAINMENT ISOLATION VALVES"

JPTS-89-027

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

Attachment II

SAFETY EVALUATION

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed change to the James A. FitzPatrick Technical Specifications deletes Table 3.7-1 (entitled, "Primary Containment Isolation Valves" on pages 198 through 209) along with the associated notes and any references to this table. References to a similar table in the updated FitzPatrick Final Safety Analysis Report (FSAR) replaces the deleted table.

Specifically, this amendment proposes the following changes to the FitzPatrick Technical Specifications:

1. Table 3.7-1 and its associated notes currently occupy twelve pages extending from page 198 to 209. All twelve pages have been replaced with a single page bearing the words "PAGES 198 THROUGH 209 HAVE BEEN DELETED." A note on the bottom of the page will alert the reader that the "Next page is 210."

   These twelve pages previously contained the following:

   Pages 198 through 209 - Table 3.7-1, "Primary Containment Isolation Valves."

   Page 207 - "Notes for Table 3.7-1, Isolation Signal Codes," listed and described the isolation signal codes.

   Pages 208 and 209 - "Notes for Table 3.7-1," listed fifteen explanatory notes associated with Table 3.7-1.

2. In Section 3.2 on page 55 the phrase, "Details of valve grouping and required closing times are given in Specification 3.7" has been revised to read, "Details of valve grouping are given in the JAF FSAR section 7.3." The phrase "assuming a 60 second valve closing time" and the sentence following the phrase have both been deleted.

3. In Section 3.2 on page 56 the phrase, "See Specification 3.7 for isolation valve closure group" has been revised to read, "Details of the isolation valve group are given in the JAF FSAR section 7.3." The phrase "it causes isolation of Groups B and 3" has been replaced with "it causes isolation of Groups B and C."

4. Table 3.7-1 has been deleted from the List of Tables on page vi.

5. In Section 3.7.D.1 on page 185, the phrase, "specified in Table 3.7-1" has been deleted. The next sentence has been deleted.

6. In Section 3.7.D.2 on page 186, the phrase "listed in Table 3.7-1" has been replaced with "listed in the JAF FSAR section 7.3."

7. In Section 4.7.D.1.a on page 185, the phrase "and for closure times as specified in Table 3.7-1" has been revised to read, "and for closure times as specified in the JAF FSAR section 7.3"

8. Section 4.7.D.2 on page 186, the phrase "listed in Table 3.7-1" has been replaced with "listed in the JAF FSAR section 7.3."

9. A reference to the table of containment isolation valves in the updated JAF FSAR section 7.3 has been added to Bases 3.7 and 4.7 (on pages 192 and 197 respectively).

II. PURPOSE OF THE PROPOSED CHANGES

The purpose of this change is to reduce the administrative resources required by both the NRC and the Authority to maintain an accurate and up-to-date table of containment isolation valves. The elimination of lists of components has been identified as a generic improvement by both industry and NRC programs.

The Authority endorsed technical specification reform activities in November 1985 (Reference 2). Over four years ago, both the NRC's Technical Specifications Improvement Project Report (Reference 3) and the Atomic Industrial Forum's Report (Reference 4) endorsed the idea of using the FSAR as an appropriate place for this type of information.

Rather than preparing and submitting a Technical Specification amendment request for each Table 3.7-1 alteration, the Authority will maintain the table of containment isolation valves in the FitzPatrick updated FSAR. This will assure that the table is periodically updated without the administrative burden of an operating license amendment.

FSAR Table 7.3.1, entitled "Process Piping Penetrating Primary Containment," was included in the FSAR update issued in July 1986 (Reference 1). This new table arranges entries by containment penetration, better describes the penetrations' function and clearly identifies the associated containment isolation valve(s). Isolation signals, valve closure time, and normal status are included on the FSAR version of this table. The new table also has a note section similar to the one in the Technical Specifications, but the FSAR notes have been clarified.

Other improvements make the table more useful and easier to use. A different format eliminates the need to reduce the table photographically and unnecessary columns in the table have been deleted.

Table 4.7-2 entitled, "Exception to Type C Tests," (pages 211, 212, 213, 213a, and 213b) lists for Type C tests exceptions for certain containment penetrations and containment isolation valves. This table is not deleted as part of this amendment application and will be retained as part of the Technical Specifications.

III. EFFECT OF THE PROPOSED CHANGES

The deletion of the table of containment isolation valves is purely administrative in nature and will not degrade safety. This amendment does not alter or remove any operability or surveillance requirements currently in the FitzPatrick Technical Specifications.

10 CFR 50 contains adequate requirements applicable to containment isolation valves to assure the safe operation of the FitzPatrick plant whether or not a list of containment isolation valves is included in the Technical Specifications.

Both 10 CFR 50.59, "Changes, tests and experiments" and 10 CFR 50.71(e), "Maintenance of records, making of reports" already contain provisions which require the Authority to inform the NRC of changes to the plant. 10 CFR 50.59 permits the Authority to:

> "(i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report,...without prior Commission approval unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question."

10 CFR 50.59 requires that the Authority annually submit to the NRC "a report containing a brief description of such changes, tests or experiments, including a summary of the safety evaluation of each." This part also requires that the Authority complete a safety evaluation for each change to assure that the change does not involve an "unreviewed safety question." 10 CFR 50.59 (a) (2) defines three criteria to be applied to determine if a change "involves an unreviewed safety question."

10 CFR 50.71(e) requires the Authority to revise the FitzPatrick FSAR each year "to assure that the information included in the FSAR contains the latest material developed."

10 CFR 50.59 and 10 CFR 50.71 provide adequate assurance that changes to the plant that result in changes to this table will be evaluated by the Authority and reported to the NRC.

Two other power reactor licensees have been granted similar amendments by the NRC to delete the table of containment isolation valves from Technical Specifications (References 5 and 6).

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the James A. FitzPatrick Nuclear Power Plant in accordance with this proposed amendment would not involve a significant hazards consideration, as defined in 10 CFR 50.92, since the proposed changes would not:

1. involve a significant increase in the probability of an accident or consequence previously evaluated. The relocation of this information from the Technical Specifications to the FSAR is purely an administrative change. It will have no effect on how the plant is maintained or operated nor does it alter the plant's design. Federal regulations 10 CFR 50.59 and 10 CFR 50.71 already contain provisions that require the Authority to complete a safety evaluation of any changes to the plant, to report these changes annually, and to update the FSAR.

2. create the possibility of a new or different kind of accident from those previously evaluated. The relocation of the containment isolation valve table does not involve a modification to the plant or a change in the procedures used for plant operation.

3. involve a significant reduction in the margin of safety. A similar table has been provided in the updated FitzPatrick FSAR. The FSAR is revised in accordance with the provisions of 10 CFR 50.71(e). This amendment does not alter any operability or surveillance requirements currently in the FitzPatrick Technical Specifications.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not impact the ALARA or Fire Protection Programs at the FitzPatrick plant, nor will the changes impact the environment.

VI. CONCLUSION

These changes, as proposed, do not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

a. will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

b. will not increase the possibility for an accident or malfunction of a different type from any evaluated previously in the safety analysis report;

c. will not reduce the margin of safety as defined in the basis for any technical specification; and

d. involve no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. NYPA letter, J.C. Brons to H.R. Denton dated July 22, 1986 (JPN-86-032) regarding 1986 annual FSAR update.

2. NYPA letter, J.C. Brons to H.R. Denton dated November 15, 1985 (JPN-85-083, IPN-85-060) regarding NYPA endorsement of technical specification reform activities.

3. NRC Technical Specifications Improvement Project Final Report, "Recommendations for Improving Technical Specifications," dated September 30, 1985.

4. Atomic Industrial Forum, "Technical Specifications Improvements," Subcommittee on Technical Specification Improvements, dated October 1, 1985.

5. NRC letter, T.V. Wambach to D.C. Shelton, dated April 13, 1990, regarding the deletion of Table 3.6.2, Containment Isolation Valves in its entirety from the Davis-Besse Technical Specifications.

6. NRC letter, H. Silver to W.S. Wilgus, dated May 22, 1989 regarding the deletion of the Containment Isolation Valve Table from the Crystal River Unit 3 Technical Specifications.