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September 16, 1977

Docket No.: 50-333

Power Authority of the State
of New York
ATTN: Mr. George T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

Gentlemen:

The Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. DPR-59 for the FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application for amendment submitted by letter dated August 31, 1977, and staff discussions.

The amendment revises the Technical Specification provisions with respect to the schedule for installation and removal of a neutron flux monitor.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 29
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Const. / Provide ... Safety Evaluation ... written verification ... installation ... That we have ... were performed

GD

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DATE	9/14/77	9/14/77	9/14/77	9/15/77	9/ /77	9/16/77

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Power Authority of the State
of New York

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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. 29
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (the licensee) dated August 31, 1977, 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. 29 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald B. Zwetzig for
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 16, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace page 137 of the Appendix A Technical Specifications with the attached revised page 137. The changed area of the revised page is shown by a marginal line.

3.6 (cont'd)

3. The pump in an idle reactor recirculating loop shall not be started unless the coolant in that loop is within 50°F of the reactor coolant temperature in the reactor vessel.

B. Pressurization Temperature

1. The reactor vessel head bolting studs shall not be under tension unless the temperatures of the vessel flange and the head flange are >90°F.

2. Pressurization temperature during hydrostatic testing shall be in accordance with Fig. 3.6-1.

4.6 (cont'd)

3. Prior to starting the pump in an idle recirculation loop, the temperature of the coolant in that loop shall be compared to the temperature of the reactor coolant in the reactor vessel.

4. Recirculation loop pump RTDS shall be checked daily and calibrated once/ operating cycle.

B. Pressurization Temperature

1. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel flange and head flange temperature shall be recorded.

2. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program shall, in the main, conform to 1972 Draft revision, ASTM E185. The monitor shall be installed during the 1978 refueling outage and shall be removed and tested during the next subsequent refueling outage to experimentally verify the calculated values of integrated neutron flux that are used to determine RTNDT.

The capsule withdrawal schedule shall be in accordance with the following:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 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

Introduction

By letter dated August 31, 1977, the Power Authority of the State of New York (the licensee) requested an amendment to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant (the facility). The amendment would modify the Technical Specifications with respect to the schedule for installation and removal of a neutron flux monitor.

Background

Facility Technical Specification 4.6.B.2 presently requires that a neutron flux monitor be installed in the reactor vessel adjacent to the vessel wall prior to initial operation and that the monitor be removed at the first refueling and tested to experimentally verify the calculated value of the integrated neutron flux. During the current (first) refueling outage the neutron flux monitor referred to in this Specification was lost within the reactor vessel in the course of operations directed toward its removal. Despite extensive efforts by the licensee to locate the lost monitor, the monitor has not been found or recovered. This revision of the Technical Specifications is therefore needed to permit operation following the first refueling when the monitor has not been removed for testing and to permit installation of a replacement monitor at a future refueling.

Discussion

A. Neutron Dosimetry Considerations

The purpose of installing a neutron flux monitor is to obtain experimental neutron fluence values which can be used to verify or correct calculated neutron fluences. Such data are needed to accurately estimate the neutron exposure of the reactor vessel and, based on this, adjust reactor coolant system operating limits to reflect changes in the vessel's material properties.

Normally a neutron flux monitor is installed in the reactor vessel prior to reactor operation because the absence of radiation at that time simplifies installation. The monitor is then removed at the first refueling outage (after one operating cycle) and tested to obtain the experimental data which is sought. This exposure of the monitor, however, need not be accomplished during the first operating cycle. Rather, it may be done during any operating cycle prior to the accumulation of significant exposure of the reactor vessel ($>5x10^{17}$ nvt, E>1 Mev). The licensee states that for the FitzPatrick reactor it would take approximately 10 years to accumulate such an exposure. It is on this basis that he proposes that a replacement neutron flux monitor can be installed within the reactor vessel to provide the needed data and that this installation and removal can be completed any time during the first 10 years of operation.

We agree that a replacement neutron flux monitor can be installed to provide the needed data. We do not agree, however, that the evaluation of the experimental flux data can be postponed to the tenth year of reactor operation. Rather, we believe it should be obtained as early in the reactor operating history as practicable. At the present time the reactor vessel has been closed preparatory for operation in Cycle Two. Accordingly, even though the licensee has obtained a replacement neutron flux monitor it is not practicable to install the monitor for the present operating cycle without incurring a substantial delay in the restart of the facility. We do not believe such a delay is necessary to protect the health and safety of the public, nor do we believe it would be in the public interest. We do believe, however, that the neutron flux monitor should be installed at the end of Cycle Two and removed and tested at the end of Cycle Three. Accordingly, we have revised the licensee's proposed amended Technical Specifications to reflect this consideration. This revision has been discussed with and found acceptable by the licensee.

B. Loose Parts Considerations

We have evaluated the licensee's analysis of the safety significance of the neutron dosimeter as a loose part in the vessel during reactor operation. The concerns, as in the case of any loose part, fall into three categories:

1. Direct mechanical and/or chemical damage to the reactor vessel and internals,
2. Interference with control rod movement, and
3. Flow blockage.

The neutron flux monitor constituting the loose part in this case, consists of a 0.5 inch diameter by 6.5 inch long piece of Type 304 stainless steel tube, with 0.25 inch thick end plugs and a 0.188 inch diameter Type 304 stainless steel handle. Three loops each of copper and iron wire are contained within the 0.5 inch tube, and the overall dimensions of the monitor are 0.5" x 7" x 4 1/8". Mechanical damage to the reactor vessel walls or internals could result from wear-type damage from the monitor if it was lodged in one location but vibrating from the coolant flow, or from impact damage from the monitor if swept along by coolant flow. Vibrational wear or impact damage to the control rod mechanisms resulting in impaired shutdown or scram capability is unlikely, but the surveillance program required by the Technical Specifications and outlined below will detect such impairment and appropriate actions will then be taken. Vibrational wear or impact damage to a fuel bundle channel which could produce a reduction in bundle flow would not have consequences significantly different from those due to flow blockages which are also discussed below. In either case, impact damage is not likely because the coolant flow velocities within the reactor where the monitor could be are not large enough to impart sufficient kinetic energy to the monitor.

Mechanical damage would possibly occur in the recirculating pump and piping external to the reactor, as for instance to the pump impeller, but this is not a threat to the pressure boundary. The small size and mass of the monitor would be unlikely to cause breaching of the pump casing. Further consideration of the effect on flow is discussed later.

The presence of the neutron flux monitor in the reactor vessel raises the question as to the possibility of corrosion or other chemical effects on reactor components. In this regard, the monitor may remain intact, or may be present in broken pieces, in the reactor vessel.

If the monitor remains intact, only negligible corrosion of the Type 304 stainless steel surfaces of the monitor will occur (at the rate of <3 mils in 40 years), with no detectable corrosion product release to the reactor coolant, vessel or piping walls or to the exterior.

If the monitor tube is broken, the iron (Fe) and copper (Cu) wires within will be exposed to the reactor water. During normal operating conditions (550°F, 0.2ppm O_2 water), the Fe will corrode slowly (<3 mils/40 years) producing an adherent layer of protective magnetite (Fe_3O_4) which will be indistinguishable from carbon steel corrosion already occurring in the system. The rate of corrosion of the copper will be extremely high (>>3000 mils/40 years as based on 90Cu-10Ni at 400°F). The resulting corrosion may "plate out" in the BWR system as a function of the degree of damage to the dosimeter, its location in the reactor pressure vessel, the coolant flow rate and the clean-up system efficiency. However, the relative amount of Cu present compared to the potential surface area on which plating may occur is small. Neither the stainless steel, the iron, nor the copper would be expected to contribute to the corrosion of reactor materials.

Based on the foregoing, we conclude that corrosion or other chemical action would have no effect on the ability of the reactor components to perform their design functions for the life time of the reactor.

If the monitor fell into a fuel support grating, it could find its way into a control rod guide tube and possibly interfere with the velocity limiter and the control blade. By letter dated September 16, 1977, the licensee stated that all control blades and drives were tested for friction, scram times, and coupling before final assembly of the reactor. All control rods functioned properly which indicates that the monitor has not entered one of these areas and is not impairing rod movement. Technical Specifications require that all control rods be exercised and periodically scram tested during operation. Therefore, these tests will indicate if the monitor has moved into an area during reactor operation where it could impair rod movement or scram. Should a control rod drive or blade be found inoperable, or operating in a degraded manner, or incapable of scram, the Technical Specifications require that the blade be fully inserted and disabled. (Under some circumstances the control blade is not required to be inserted, but only if it can be shown that adequate shutdown margin is maintained with the inoperable rod stuck plus the operable rod with the greatest reactivity worth stuck in the fully withdrawn position). Because of this surveillance, and these operating limits the lost monitor does not endanger the shutdown and the scram reliability of the reactor.

The presence of the lost neutron flux monitor could cause flow blockage to occur within the recirculating pump (locked rotor), within a jet pump nozzle, or at the orifice of a fuel assembly.

Recirculating pump seizure is a relatively mild transient for this facility and is fully addressed in the Final Safety Analysis Report (FSAR). Based on the FSAR analyses, the consequences of a recirculating pump siezure are well within safety limits.

Blockage of a jet pump would result in a relatively mild readjustment in core flow and is well bounded by the recirculating pump trip and seizure analyses in the FSAR. In addition, such blockage would be easily detected by jet pump instrumentation. Therefore, jet pump blockage would not result in a violation of a safety limit.

The possibility that the lost monitor could cause blockage of flow to a fuel bundle and consequently induce fuel damage has been evaluated.

The maximum flow blockage which could occur should the monitor remain intact would be 30% for a central bundle or 45% for a peripheral bundle. For such blockages fuel damage is possible only if an operational transient should also occur and thereby reduce the critical power ratio (CPR) of the blocked assembly to the point of transition boiling. This would be possible should the steady state CPR of the flow blocked assembly be significantly lower than the intended operating limit Minimum Critical Power Ratio (MCPR) for the plant during the upcoming cycle. However, flow blockage not only reduces flow, but also increases voids and consequently reduces assembly power. For blockage from an intact monitor, the licensee states that the reduction in steady state CPR due to the flow decrease is offset by the increase in CPR due to the bundle power decrease. Therefore, the steady state CPR of the flow blocked assembly will be approximately the same as for the normal assembly and, thus, should be sufficiently above the safety limit MCPR to assure that transition boiling will not occur due to an operational transient.

Only if the dosimeter breaks into several pieces all of which eventually block the same bundle orifice could a flow reduction sufficient to produce melting and dispersal of fuel occur. This is a very low probability event but should such blockage occur the resultant activity release due to fuel cladding failure would be detected in the main steamline radiation monitors which would scram and isolate the reactor. No damage to adjacent bundles would occur and the resultant offsite doses would be less than 10 CFR 20 limits.

Based on the foregoing, we conclude: (1) that the proposed change in the the schedule for installation and removal of the neutron flux monitor, as revised per discussion with the licensee, does not reduce the safety margin for operation and is therefore acceptable, (2) that the loose part (neutron flux monitor) will not cause mechanical or chemical damage to reactor components that would affect their functions and therefore would not cause a decrease in safety margin and an increase in the probability of an accident, or the consequences of an accident, (3) that the loose part does not affect the shutdown or scram reliability of the reactor and therefore reduce the safety margin; and (4) that the probability of occurrence of a condition of flow blockage which could lead to fuel damage is highly unlikely and, therefore, that release of radioactivity as a result of fuel damage is extremely improbable. Further, even if such fuel damage were to occur, it would necessarily be limited to a very small fraction of the core and the consequences would be much less than those already analyzed.

Therefore, the proposed Technical Specification change is acceptable, and it is acceptable to operate the reactor with the loose part (the neutron flux monitor) within the vessel.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 16, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 29 to Facility Operating License No. DPR-59, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility) located in Oswego County, New York. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specification provisions with respect to the schedule for installation and removal of a neutron flux monitor.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment submitted by letter dated August 31, 1977, (2) Amendment No. 29 to License No. DPR-59, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Oswego County Office Building, 46 E. Bridge Street, Oswego, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16th day of September 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald B. Zwetzig

Gerald B. Zwetzig, Acting Chief
Operating Reactors Branch#4
Division of Operating Reactors