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SUPPLEMENT NO. 2 TO THE SAFETY EVALUATION OF THE

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

U. S. ATOMIC ENERGY COMMISSION
DIRECTORATE OF LICENSING
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Supplement No. 2

To The

Safety Evaluation of the

James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333

U. S. Atomic Energy Commission

Directorate of Licensing

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I. Introduction

The Safety Evaluation Report (SER) for the James A. FitzPatrick Nuclear Power Plant was issued on November 20, 1972. Supplement No. 1 was issued on February 1, 1973 to provide additional information on previously unresolved issues such as anticipated transients without scram, inservice inspection and offsite power and to discuss the ACRS recommendations. Several items (such as the control rod drop accident, high energy line breaks outside the containment and the recirculation pump overspeed protection) were still under study and evaluation at the time Supplement No. 1 was issued and are listed as revised sections. Other information is listed as new sections, which have been finalized since the issuance of Supplement 1.

The purpose of this supplement is to update the SER and to discuss the resolution of all remaining open items. This supplement contains the following information:

1. New Sections
    a. Fuel Densification
    b. Radioactive Materials Safety
    c. Quality Assurance Independence
    d. Main Steam Line Leakage Control System
    e. Cask Drop Protection
    f. Reactor Pedestal Repair

2. Revised Sections From Previously Outstanding Items

   a. Environmental Qualification Tests

   b. Control Rod Drop Accident

   c. Seismic Design Criteria

   d. Fracture Toughness

   e. Reactor Recirculation System

   f. Rod Sequence Control System

   g. High Energy Line Head Outside Containment

   h. Primary Containment Leakage Test Control

The subjects addressed herein are numbered to coincide with the applicable sections of the SER. An updated chronology is presented in Appendix A.

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II. Revised Sections of Safety Evaluation Report

3.0 REACTOR

3.4 Reactivity Control

3.4.1 RSCS Group Notch Modification

The RSCS for FitzPatrick has been previously described in Supplement 1 to the SER. FitzPatrick will now utilize the Group Notch RSCS as described in PASNY letter dated May 3, 1974. The staff evaluation of this system was done on a generic basis and is contained in the staff document, Review of Rod Sequence Control Systems with Group Notch (RSCS/BWR-4) attached to memorandum from V. Stello to V. Moore, dated June 6, 1974. Although installation of this system was not required before the first fuel loading PASNY has chosen to install this modification before licensing.

At low-power levels, below 20%, the RSCS forces adherence to acceptable rod patterns and a rod worth limit to preclude unacceptable consequences in the event of a control rod drop. The RSCS is not required for rod pattern control above 20% of rated power. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ±10% of full power, the nominal instrument setting of 30% of rated power provides assurance that the RSCS will be operative to 20% of rated power.

The 100% to 50% rod density control is the same as previously described on FitzPatrick. This design requires specific groups to be selected and fully withdrawn to achieve a checkboard pattern at the

50% density point. The operator has the option to select one of four groups of rods. The groups are identified as A_{12}, A_{34}, B_{12}, and B_{34}. These groups are compatible with the Rod Worth Minimizer (RWM) Groups 1, 2, 3, and 4. Having selected a sequence A or B, then all the rods in that sequence must be withdrawn to the full limit before rods in the other sequence can be moved.

The full in and full out limit switches used as inputs to the indicators on the full core display are used as the inputs to the sequence logic through optical isolators. Redundant full in and full out switches are used as inputs to the Rod Position Indication System (RPIS) and the RWM.

When all the rods of a selected sequence, A or B, are full out movement of the remaining rods is blocked until the selector switch is moved to the normal position. The movement of the inserted rods is then controlled by the group notch logic portion of the RSCS from the 50% density to 30% power level. The reverse order is required during shutdown.

The group notch logic system is a hard wired system designed with a combination of memory latches and assembly gates wired into groups of four, five, six, or eight rods. The system restricts motion of the selected rod to one notch only and resets the logic for any group when all the rods in that group have been moved one notch in their selected direction.

Special requirements have been placed in the Technical Specification, Section 3.3, since portions of the RSCS must be bypassed during scram time measurement tests. For testing below 20% power the RSCS and the RWM must be operable. Only the rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram time tested. The position of groups selected for withdrawal on the group notch logic portion of the RSCS, needed to reach power levels to produce reactor test conditions above 950 psig at corresponding saturated temperature, must be simulated as fully inserted. The remaining rods are tested by shutting down and reversing the startup sequence or are tested when the RSCS restraints are automatically bypassed.

The components of the RSCS will meet the quality control and assurance requirements of a single channel of the reactor protection system. The equipment added to or interfacing the Reactor Manual Control System will be qualified in conformance to IEEE Std 323-1971.

The design provides for testability of both the sequence portion and group notch portion of the system.

The staff concludes that the design of the Rod Sequence Control System with Group Notch Control (RSCS/BWR-4) is acceptable based on its making the probability of a postulated damaging control rod drop accident negligibly low.

3.6 Fuel Densification

3.6.1 Background

This discussion updates and supersedes the interim report on the subject of fuel densification previously provided in Section 4.2 of the SER and Supplement No. 1.

On January 17, 1973, General Electric (GE) submitted the topical report "Densification Considerations in BWR Fuel Design and Performance," NEDM-10735 (Ref 2) which applied to GE boiling water reactors generally. Subsequently, GE submitted five supplements (Ref 3, 4, 5, 6 and 7) to this topical report which provided additional information. Based on this information the Regulatory staff issued the report entitled "Technical Report on Densification of General Electric Reactor Fuels" (Ref 8). PASNY provided analyses of the effect of densification on steady state operations, operating transients and postulated accidents on FitzPatrick in their letter of August 20, 1973, and the referenced GE topical report "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735, Supplement 6, August 1973. Subsequently, GE has submitted two revisions to that supplement and it is now called NEDM-10735, Supplement 6, 7, 8 (Ref 9).

3.6.2 Densification Effects

A detailed discussion of the causes and effects of densification including the results of observations of irradiated fuel in both test and power reactor fuel, an investigation of the possible mechanisms

and evaluation of the controlling parameters, is presented in the staff reports on densification (Ref 1 and 8). At this time the only clear conclusion that can be drawn is that under irradiation, fuel pellets can shrink and decrease in volume with corresponding changes in pellet dimensions. Four principal effects are associated with the dimensional changes resulting from densification. A decrease in length of pellets could result in the formation of axial gaps in the column of fuel pellets within a fuel rod. Two effects are associated with axial gaps. First, if relatively large axial gaps form, creep-down of the cladding later in life may lead to collapse of the cladding into the gaps. Second, axial gaps produce a local increase in the neutron flux and generate a local power spike. A third effect, which results from a decrease in pellet length, is a directly proportional increase in linear heat generation rate.

A fourth effect, which results from a decrease in pellet radius and an increase in the pellet-clad radial clearance, is a decrease in pellet-clad thermal conductance (gap conductance). Decreased conductance would increase fuel pellet temperature and stored energy and decrease the heat transfer capability of the fuel rod. Each of these four effects has been considered in evaluating the total effect that fuel densification might have on normal operation, transients and accidents.

Based on experimental evidence that no collapse has been observed in BWR fuel rods and on the results of calculations performed

independently by the staff and GE, the Regulatory staff has concluded that typical BWR fuel will not collapse during the first cycle of operation (Section 3.4.2, Ref 8). GE has also calculated the creep collapse of fuel in later cycles using a model which includes the modifications specified by the staff (Section 3.4.2, Ref 8). The results of these calculations for FitzPatrick fuel are reported in Supplement 6, 7, 8 of the GE report (Ref 9). The calculations indicate that clad collapse will not occur in FitzPatrick Type I fuel for residence times less than 2.9 years, and in FitzPatrick Type II and III fuel for residence times less than five years. The expected residence time of the Type I fuel is less than two years. The staff has reviewed the GE calculations and performed independent calculations, which also predict that collapse will not occur. Based on the calculations and experimental evidence, the staff concludes that creep-collapse need not be considered as affecting normal operation, transients or accidents.

The increase in linear heat generation rate (LHGR) resulting from contraction of the fuel is offset by compensating factors. Although pellets with initial densities less than the mean initial density will contract more than the average pellet, such pellets also contain correspondingly less fuel and produce less power in a given neutron flux. Therefore, only contraction from an initial mean pellet density need be considered in determining the LHGR. In the case of the

FitzPatrick fuel, this contraction is offset by thermal expansion, as shown by calculations summarized in Table 3-1 of Supplement 6, 7, 8 of the GE report (Ref 9). Since the increase in fuel column length due to thermal expansion was not considered in the original design calculations or transient and accident analyses, and since the effect of thermal expansion offsets the effect of densification on LHGR, it is appropriate to use the design LHGR in the analyses of normal operation, transients and accidents when considering the effects of densification. This was done in all the analyses presented by GE in Supplement, 6, 7, 8 of the topical report (Ref 9).

Calculations by GE of power spikes resulting from possible axial gaps in the fuel take into account the peaking due to a given gap, the probability distribution of peaking due to the distribution of gaps, and the convolution of the peaking probability with the design radial power distribution. Based on an examination of the methods used, comparison with requirements and approved models given in the staff densification report, and check calculations performed for the staff by Brookhaven National Laboratory, the staff concluded in their report (Ref 8) that, if appropriate gap assumptions are made regarding sizes, the GE calculational method is acceptable. The results of calculations of power spikes using acceptable gap sizes are summarized in Figure 3-6 of Supplement 6, 7, 8 of the GE report (Ref 9). During normal operation there is a 95% confidence that no more than one rod

would have a power spike greater than approximately 4% at the top of the fuel. At the midplane the corresponding power spike would be approximately 2%. When the reactor power is low and there are no voids, the spike could be greater. Under these conditions, there is a 95% confidence that no more than one rod would have a power spike greater than 5% at the top of the fuel.

Pellet-clad thermal conductance is a function of a gap size and linear heat generation rate. The staff has reviewed the experimental data and analyses that GE has submitted to justify their correlation of gap conductance, examined the uncertainties in the data, and performed independent calculations with a fuel thermal performance computer program. The pellet-clad thermal conductance correlation used for FitzPatrick is depicted in Figure 3-10 of Supplement 6, 7, 8 of the GE report (Ref 9). It is based on experimental data and predicts with a 95% confidence that 90% of the total population of pellet-clad conductances exceed the prediction. The staff concludes that this correlation when applied for the gap size adjusted for the effects of densification is acceptable.

3.6.3 Evaluation of Effects of Densification

3.6.3.1 Normal Operation

The operating limits affected by fuel densification are the design values of maximum linear heat generation rate (LHGR) and minimum critical heat flux ratio (MCHFR). In order to maintain adequate

safety margin during normal operation, power spike considerations require that: (1) the LHGR in any rod at any axial position be less than the design value of 18.5 kw/ft by a margin equal to or greater than the power spike calculated using the accepted model; and (2) the MCHFR calculated with power spikes be maintained above the steady state design limit of 1.9. As discussed previously, this power spike penalty will assure at the 95 percent confidence level that no more than one rod will exceed the design limits.

3.6.3.2 Transient Performance

The key transients for evaluation of BWR performance are those associated with overpressurization, which might imperil the integrity of the primary coolant pressure boundary, and with reduction of coolant flow, which might imperil the integrity of the fuel clad. The transient resulting from a turbine trip without opening the bypass valves is representative of transients that might result in overpressurization. The transient resulting from the simultaneous trip of both recirculation pump drive motors is representative of transients that result in a rapid reduction of core flow.

Following isolation of a BWR, such as would result from closure of the turbine stop and bypass valves, stored and decay energy from the core increases the coolant temperature and pressure. Since densification might reduce the pellet-clad conductance and increase the stored energy, densification could effect the peak pressure following a transient. GE

has calculated the increase in heat flux, fuel temperature and peak pressure in the primary coolant system following a turbine trip transient without bypass using gap conductances as low as 400 Btu/hr-ft^2-°F (Ref 9). A conductance of 400 Btu/hr-ft^2-°F is representative of the average fuel rod and its use is appropriate since the average fuel rod stored energy is the appropriate parameter to use when evaluating coolant system pressure. GE analyses indicate that the calculated peak pressure is increased only 4 psi and is not significantly greater than the system pressure calculated using the value of 1000 Btu/hr-ft^2-°F for gap conductance. Using a conductance of 400 Btu/hr-ft^2-°F increased the calculated fuel temperature 13°F and the heat flux 1%. These increases are also insignificant.

Following a rapid reduction in core flow, such as would result from simultaneously tripping both recirculation pump motors, the MCHFR will decrease. A MCHFR of 1.0 is taken as a design limit for fuel damage. The slower thermal response of rods with densified fuel can result in a lower MCHFR following a rapid flow reduction. GE has calculated that the MCHFR would be reduced no more than 0.07 including the effects of fuel densification. This would reduce the FSAR value of 1.24 for FitzPatrick to no less than 1.17.

Based on the results of the GE calculations, the staff concludes that changes in gap conductance resulting from fuel densification would affect the course of flow and pressure transients. However, pressure or MCHFR limits would not be exceeded.

3.6.3.3 Accidents

3.6.3.3.1 Refueling Accident

Since fuel densification does not affect any parameters used in the evaluation of the refueling accident, the consequences of this accident are not changed.

3.6.3.3.2 Control Rod Drop Accident

A generic evaluation by the staff of the control rod drop accident has been underway for the past several months. GE has submitted topical reports revising the techniques for analyses of the control rod drop accident including among other features, a change in the method for modeling the rate of negative reactivity insertion. These topical reports and revised analyses are under review. The safety evaluation of the rod drop accident without the effects of densification is presented in the FitzPatrick SER. The parameters affected by densification are: initial stored energy and heat generation and heat transfer during the transient. Other parameters important to the analysis such as gross power distribution; delayed neutron fraction; and the reactivity changes produced by the dropped rod, the scram insertion of the other rods and Doppler feedback are not significantly affected by densification.

At low initial power the only effect of densification that is important to the analysis of the rod drop accident is the local perturbation of the power distribution resulting from axial gaps in the column of fuel pellets. This power spiking effect would be very localized and

affect only a few fuel rods. The peak enthalpy will occur in the upper region of the core and, as discussed previously, the magnitude of the power spike will be less than 5% even at the top of the core.

The Rod Sequence Control System (RSCS) is designed to preclude the movement of out of sequence control rods below 20% reactor power. The peak enthalpy resulting from the dropping of an in-sequence control rod with the maximum worth is calculated to be never greater than 230 cal/g at low initial power. Therefore, even if the dropped rod were in a region with a 5% power spike, the calculated peak fuel enthalpy would be within the 280 cal/g design limit.

At high initial power the effect of densification that is important to the analysis is the higher initial stored energy due to decreased gap conductance. GE has submitted parametric results on the effect of reduced gap conductance on the initial stored energy for the control rod drop accident (Ref. 10). Reference 10 shows that in the worst case bundle the maximum initial stored energy at full power, with reduced gap conductance due to densification, is typically 167 calories per gram. Based on low power rod drop accident calculations, GE estimates that the maximum energy added due to the drop of a control rod is only 50 calories per gram. However, the 50 calories per gram does not occur in the same bundle as that with the peak initial enthalpy. Therefore, the peak fuel enthalpy of a bundle could not exceed 217 calories per gram.

An independent calculation performed by the Brookhaven National Laboratory for the Regulatory staff indicates that the maximum energy

added due to the drop of a control rod at high initial power is less than 75 calories per gram, and it does not occur in the same bundle as the peak initial enthalpy. Thus, based on the staff calculations the peak enthalpy of any fuel rod is less than 242 calories per gram. This is well below the 280 calories per gram limit.

The radiological consequences of the rod drop accident depend on the number of fuel rods that might suffer cladding damage as a result of the accident. GE estimates that at low initial power and without densification effects the rod drop accident will result in fewer than 600 damaged fuel rods. Similar calculations were not provided for the rod drop accident at high initial power. Based on the available information the staff can not conclude that the number of damaged fuel rods at high initial power is fewer than the corresponding number calculated for the low initial power accident. Consequently, we have extended our generic review of the BWR rod drop accident to include the high initial power case.

We have requested GE to furnish the necessary information for our use in this generic evaluation. Pending receipt of this information, we have determined the number of fuel rods that would have to fail before the guideline doses of 10 CFR Part 100 are exceeded. We find that if ten times the GE estimated fuel rods were to experience clad perforations due to a rod drop accident while operating at high power levels, the resulting 24 hour dose at the low population zone distance would be approximately 20 rem-whole body and less than 20 rem-thyroid.

While the staff is not able to conclude, at this time, that the number of rod perforations for this postulated accident (rod drop of high power levels) is less than the 600 estimated by GE, we can, however, conclude that it would not be anywhere near the 6000 rods cited above. We, therefore, conclude that, although we are continuing our evaluation of the matter, the dose consequences of the rod drop accident, were it to occur while operating at high power levels, are within the 10 CFR Part 100 criteria and that the peak enthalpy is less than our acceptance limit of 280 cal/gm. These consequences are, therefore, acceptable.

3.6.3.3.3 Main Steam Line Break Accident

As in the analysis of transients, the effect of reduced gap conductance resulting from densification is an increase in stored energy and transient heat flux. However, calculations demonstrate that the expected reduction in conductance for this plant does not result in departure from nucleate boiling during the transient (Ref. 9). As in the GE calculation presented in the FSAR (gap conductance equal 1000 Btu/hr-ft^2 - °F) no clad heatup is predicted to occur and consequently the main steam line break accident is unaffected by densification.

3.6.3.3.4 Loss-of-Coolant Accident

3.6.3.3.4.1 Small Break LOCA

As in the analysis of a transient, the effect of reduced gap conductance resulting from densification is an increase in stored energy

and transient heat flux. A higher initial stored energy, when transferred to the coolant during blowdown, maintains the pressure, and increases the break flow rate resulting in a quicker actuation of the Automatic Depressurization System. Therefore, the reactor is depressurized sooner and the low pressure emergency core cooling systems refill the vessel sooner. Since all stored energy is removed during the initial phase of the blowdown, only the decay heat, which is the same in both cases, affects the clad temperature. The net effect, as the GE calculations (Ref. 9) demonstrates it, is a small change in peak clad temperature. Therefore, densification does not have an important effect on the small pipe break accident.

3.6.3.3.4.2 Design Basis LOCA

Following a postulated break of a recirculation pipe, densification can affect the hydraulic response of the reactor as calculated by the blowdown analysis and the thermal response of the fuel as calculated by the heatup model. The effect on the blowdown is much less significant than the effect during the heatup.

As discussed in the review of the transient analysis, the effect of densification is a reduction of gap conductance and a corresponding increase in stored energy and transient heat flux. The increased energy and heat flux result in a slightly modified hydraulic response following the LOCA. However, as shown in Figures 4-7 and 4-8 of Supplement 6, 7, and 8 to the GE report (Ref. 9), the flow rates are

not significantly changed and the time of departure from nucleate boiling is unchanged. Therefore, the convective heat transfer coefficients are not significantly changed as a result of densification.

The heatup of the fuel is, however, significantly changed primarily as a result of increased stored energy. Although the formation of axial gaps might produce a local power spike, as discussed previously the spike would be approximately 2% at the axial midplane. As discussed in the staff report (Section 4.3, Ref. 8), it is improbable that more than one spike of significant magnitude would occur at any axial elevation and that a 1% power spike would result in only a 4°F increase in peak clad temperature. Therefore, the effect of power spikes can be neglected in the heatup analysis.

The peak clad temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate and stored energy of all the rods in a fuel assembly at the axial location corresponding to the peak of the axial power distribution. The stored energy is dependent on the LHGR and the pellet-clad thermal conductance. As discussed, the conductance is based on a correlation which underpredicts 90% of the data with a 95% confidence for a selected gap size. The gap size is calculated as specified in the AEC Fuel Densification Model assuming that the pellet densified from the initial density to 96.5% of theoretical density. Since peak clad temperature is primarily a function of average stored energy, the

density of 48 rods is taken as the two standard deviation lower bound on the measured initial "boat" pellet density. For the most critical rod, the two standard deviation lower bound on initial density of individual pellets was assumed. The result of calculations of peak clad temperature assuming the operational limitations discussed in Reference 9 are presented in Figs. 4-9Q1, 9Q2 of Supplement 6, 7, and 8 to the GE report (Ref. 9).

Subsequently, General Electric (GE) submitted a report NEDO-20181, "GEGAP III A Model for the Prediction of Pellet-Clad Thermal Conductance in BWR Fuel Rods," November 1973, with related proprietary information provided in NEDO-20181 Supplement I (Proprietary), November 1973. The AEC Regulatory staff has reviewed the GEGAP III model and has issued the report entitled "Supplement 1 to the Techncial Report on Densification of General Electric Reactor Fuels" dated December 14, 1973. On December 5, 1973, letters were sent requesting that licensees provide the necessary analyses and other relevant data needed to determine the consequences of densification and its effect on normal operation, transients, and accidents using an enclosure, "Modified GE Model for Fuel Densification." The licensee provided an analysis of the effect of densification on normal operations, transients and accidents for the James A. FitzPatrick Nuclear Power Station in their letter of July 10, 1974, and referenced the GE letter, "Plant Evaluations with GEGAP III," dated December 12, 1973.

The "Modified GE Model for Fuel Densification" results in an increase in the pellet-clad thermal conductance which is different than the instantaneous conductance. This results in a decrease in the stored energy of the fuel rods. The pellet-clad thermal conductance value lies between the value used in the FSAR and the value used in the Regulatory staff's technical report of August 23, 1973. The results of using the gap conductance from the modified version of GEGAP III in the analysis of normal operation and transients is to produce results between those evaluated in the FSAR and those used in the Regulatory staff's technical report of August 23, 1973. Therefore, it is concluded that the change has essentially no effect on normal operation, and improves the margins to pressure and minimum critical heat flux ratio limits for overpressurization and core flow reduction transients.

The increase in pellet-clad thermal conductance would reduce the consequences of the design basis loss-of-coolant accident assuming a constant linear heat generation rate. The reduction would occur during the heatup phase of the accident as a result of the decreased initial stored energy. However, the stored energy is also dependent on the linear heat generation rate of the fuel. A reduction in stored energy then allows a compensating increase in linear heat generation rate, such that operating flexibility is increased while compliance with the Interim Acceptance Criteria is still maintained. The limit curves for MAPLHGR specified in this change represent the most limiting of three

limits: MCHFR, cladding strain, and peak clad temperature following LOCA. The Regulatory staff concludes that the limitations of the average linear heat generation rate of all rods in any fuel assembly at any axial location to the values given in Figures 3.5.1 and 3.5.2 of Specification 3.5.H and 3.5.I of the James A. FitzPatrick Technical Specifications will assure that the calculated peak clad temperatures will not exceed 2300°F.

The staff concludes that limitation of the average linear heat generation rate of all the rods in any fuel assembly at any axial location to the values of the curves labeled ω will assure that calculated peak clad temperatures for the design basis LOCA will not exceed the 2300°F limit.

3.6.4 Conclusions

The Regulatory staff has reviewed the postulated effects of fuel densification on FitzPatrick for normal operations, transients and accidents. The staff has concluded that Technical Specifications limits are required to ascertain that even with the postulated effects of densification: (1) the 18.5 kw/ft design value of LHGR will not be exceeded; (2) the stated steady state design limit of 1.9 MCHFR will not be exceeded; and (3) the maximum average planar LHGR will not exceed the ω curves of Figures 3.5.1 and 3.5.2 in the FitzPatrick Technical Specifications. If these limits are met, there is reasonable assurance that operation of FitzPatrick at power levels up to and including 2381 MWt will not result in undue risk to the health and safety of the public.

3.6.5 References

1. "Technical Report on Densification of Light-Water Reactor Fuels," by the USAEC Regulatory staff, November 14, 1972.
2. "Densification Considerations in BWR Fuel Design and Performance," NEDM-10735, December 1972.
3. "Response to AEC Questions - NEDM-10735," NEDM-10735, Supplement 1, April 1973.
4. "Response to AEC Questions - NEDM-10735, Supplement 1," NEDM-10735, Supplement 2, May 1973.
5. "Responses to AEC Questions - NEDM-10735, Supplement 1," NEDM-10735, Supplement 3, June 1973.
6. Responses to AEC Questions - NEDM-10735," NEDM-10735, Supplement 4, July 1973.
7. "Densification Considerations in BWR Fuel," NEDM-10735, Supplement 5, July 1973.
8. "Technical Report on Densification of General Electric Reactor Fuels," August 23, 1973, Regulatory staff, USAEC.
9. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," NEDM-10735, Supplement 6, 7, 8, August 1973, General Electric Co.
10. Letter, J. A. Hinds to V. Stello, subject: "Fuel Densification," dated August 30, 1973.

11. Letter, J. A. Hinds to V. Stello, subject: "Fuel Densification," dated October 2, 1973.

12. Letter, I. F. Stuart to V. Moore, subject: "Fuel Densification," dated October 25, 1973.

4.0 REACTOR COOLANT SYSTEM

4.3 Fracture Toughness

PASNY has established and submitted, in Supplement 20 to the FSAR, operating pressure and temperature limitations during startup, shutdown, and hydrostatic testing of the reactor coolant system. These limitations reflect the recommendations of Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda to the ASME Code, Section III.

The minimum temperature for the 1000 psig operating pressure leak test is 130°F. The minimum temperatures for the 1190 psig system pressure tests and the hydrostatic tests at 1563 psig are 150°F and 173°F, respectively. We conclude that these temperature and pressure limitations are acceptable.

4.4 Reactor Recirculation System

The following information is an update of that contained in Section 4.4 of the SER and Supplement 1.

In Supplement 20 to the FSAR, PASNY presented the results of a study to determine the effects of missile generation from recirculation pump overspeed following a LOCA. The study addressed the probability of destructive pump overspeed resulting in escape of high energy missiles from the piping system with these missiles causing damage to the primary containment or internal piping.

The referenced study indicates that the probability of a recirculation pump missile ejecting from a broken recirculation pipe and causing loss of containment integrity is about 3×10^{-8}. The staff concludes that the probability of this potentially damaging event is sufficiently small, i.e., within a safety objective for the probability of a particular potential failure path of about 10^{-7} per year.

We conclude that the applicant's proposal to provide a decoupling device and not to add pipe restraints is acceptable. Our conclusions are based on the following:

1. The decoupling device between the recirculation pump and motor will prevent the formation of motor assembly missiles.
2. The staff's evaluation of the applicant's probability analysis indicates the concurrent probability of fuel damage and recirculation pump missile ejection causing loss of containment integrity is acceptably small (i.e., less than 1×10^{-7} per year). Accordingly, the delays and risks associated with installing additional pipe restraints outweigh the incremental small gain in safety.
3. The applicant had previously agreed to install decouplers between the pumps and motors to prevent excessive motor overspeed. These were to be installed at the first refueling outage. However, the

General Electric Company is currently having difficulty in procuring and design testing of a prototype and has indicated that the installation of the decoupler may be extended to the second refueling. The staff finds that because of the low probability of the event, the addition of even another 2 years delay would not significantly affect the staff's conclusion.

5.0 CONTAINMENT SYSTEMS

5.2 Primary Containment

5.2.4 Leakage Test Program

This information supplements the commitment made by the applicant in Section 5.2.4 of the SER. PASNY has presented additional information on the drywell-to-suppression chamber vacuum breaker valves, including an analysis of allowable bypass leakage and the details of the surveillance testing program in FSAR Supplements 19 and 20. The leakage capacity detectable by testing is approximately 10% of the allowable bypass leakage which corresponded to 1/10 the allowable leakage using a 1" orifice.

The leakage of 71 scfm from the drywell that could bypass the suppression chamber, which is approximately 10% of the allowable, will be part of the Technical Specifications. Position lights will be installed to indicate the position of the vacuum breaker valves so that it can be determined if all valves are in the fully closed position. The surveillance requirements for the allowable leak rate corresponding to a leak rate less than 0.25 in water/min over a ten minute period with the drywell at one psid conducted once per operating cycle, will be part of the Technical Specifications, Section 3.7.A.5.

Based upon our evaluation of PASNY's information we conclude that the analysis is acceptable and that the planned leakage testing together with the vacuum breaker surveillance operability tests are adequate to assure that leakage that could bypass the suppression pool is within acceptable limits.

7.0 INSTRUMENTATION, CONTROLS AND ELECTRICAL SYSTEMS

7.6 Radiation and Environmental Qualification

7.6.2 Environmental Qualification Tests

This information is an update of that contained in Section 7.6.2 of the SER and Supplement 1.

Environmental qualifications testing of the safety/relief valve solenoids was in progress at the time Supplement No. 1 to the Safety Evaluation was prepared. In Supplement No. 21 to the FSAR, PASNY stated that the solenoid valves failed the qualification test. Replacement solenoid valves from a different vendor have passed the qualification test environments as specified in the FSAR and the existing valves will be replaced with the qualified valves prior to fuel loading. The staff considers this acceptable.

8.0 AUXILIARY SYSTEMS

8.4 Fuel Handling and Storage

The safety evaluation considered the capability of the spent fuel storage pool to withstand an inadvertent drop of the shipping cask. The applicant has provided administrative and design features which will provide adequate protection against a straight vertical drop. However, the staff review revealed that the present system of mechanical and administrative devices would not provide adequate protection against the occurrence or consequences of angle drops which could occur assuming a single active mechanical failure.

We have required that PASNY select, demonstrate, and implement suitable means for protecting the pool against all credible cask drops, or for precluding unacceptable cask drops resulting from any single active failure. This system shall be functional at the time of the first refueling.

PASNY has undertaken an investigation of several preventive systems and concepts considered by other plants including: the redundant trolley, the dashpot damper, safety cables, an extended crushable honeycomb pad and a hydraulic cylinder operated elevator-like platform. While a specific system has not yet been selected, PASNY has committed to the incorporation of additional protection to mitigate the consequences of the postulated cask tilt and drop. The details will be presented to the AEC for evaluation and approval on a time scale to

permit the system to be operational at the time of the first refueling outage.

8.8 Main Steam Line Leakage Control System

In its construction report letter of December 15, 1972 on the FitzPatrick Plant the Advisory Committee on Reactor Safeguards noted that additional information should be provided by the applicant regarding leakage through the main steam line isolation valves. The ACRS asked that evaluation should be made of the operating experience with such valves for FitzPatrick and other reactors, and that, if appropriate, suitable changes be made in the FitzPatrick Plant to maintain acceptably low leakage rates.

After a review of the FitzPatrick data for a main steam line break the staff determined that FitzPatrick should install a system to limit the leakage through the main steam lines to collect low pressure leakage and process it through the standby gas treatment system. This concept has been found acceptable in principle. The design criteria, which included the system to be single failure proof, the radiological release limits to be below 10 CFR 100, and the system to start operation within 20 minutes from the start of the accident, were sent to PASNY by our letter dated May 4, 1973. PASNY amended their basic information with letters dated July 27 and December 4, 1973.

While a final design has not yet been approved, the basic principle of operation of the proposed leakage control system is acceptable to

the staff. The details of the final design will be presented to the staff for evaluation and approval on a time scale to permit the system to be operational at the time of the first refueling outage. PASNY has agreed to follow criteria subject to our final approval. The staff concludes that due to the relatively short period of time of operation until the first refueling that plant operation without this system will not significantly affect the health and safety of the public.

9.0 STRUCTURES AND EQUIPMENT

9.2 Structural Design Criteria

9.2.4 Seismic Design Criteria

The following information is an update of that contained in Supplement 1 of the SER.

PASNY has presented additional information, in Supplement 20 to the FSAR, regarding the reanalysis of peak resonant responses using a factor of 1.5. The analysis and evaluation of the results have been completed and show that the stresses remain within the allowable limits identified in Section 9.3 of the SER.

9.4 Reactor Pedestal

9.4.1 Background of Pedestal Crack

A crack in the upper portion of the reactor pedestal about 6 inches below the intersection of the shield wall and the top of pedestal was observed on January 2, 1973 and an initial report from the applicant, submitted February 14, 1973, gave a brief summary of the problem. PASNY submitted a more detailed special report on June 29, 1973, which contained an analysis to demonstrate that the pedestal structure was safe without the need for repair. However, evaluation by the staff concluded that adequate repair was necessary.

9.4.2 Approved Repair Method

PASNY submitted a preferred method of repairing the crack and thus restoring the reactor pedestal to its original design function. The

staff, as a result of information supplied and the commitments made by the applicant, finds that the repair proposed for the reactor pedestal, shield wall and pedestal in the region of the connection will be capable of transmitting the Category I response of the shield wall structure to the reactor pedestal. The additional structural materials added when acting in conjunction with the existing resistence capabilities will result in peak stresses at only specific locations that are within the allowables. Consequently, the staff considers the structure as repaired to be capable of fulfilling its Category I requirements.

10.0 ACCIDENT ANALYSIS

10.4 Control Rod Drop Accident

Correct last paragraph of Supplement 1, Section 10.4 to read as follows:

"The Technical Specifications will require that the RSCS be operable below 20% power level and that the control rod scram time (to 90% insertion) be less than 5.0 seconds." For a further general description of the RSCS see Section 3.4.1 of this supplement.

The final calculations on peak enthalpy and details of the rod drop accident may be found in Section 3.6 to this supplement.

10.5 High Energy Line Break Outside of Containment

A. INTRODUCTION

On December 18, 1972, and January 12, 1973, the Atomic Energy Commission's Regulatory staff sent letters to PASNY requesting a detailed design evaluation to substantiate that the design of the James A. FitzPatrick Nuclear Power Plant is adequate to withstand the effects of a postulated rupture in any high energy fluid piping system outside the primary containment, including the double-ended rupture of the largest line in the main steam and feedwater system. It was further requested that if the results of the evaluation indicated that changes in the design were necessary to assure safe plant shutdown, information on these design changes and plant modifications would be required. Criteria for conducting this evaluation

were included in the letters. A meeting was held on January 26, 1973 to discuss the information already available on the FitzPatrick Plant design concerning postulated pipe ruptures, to discuss the criteria, and to assess those areas where additional information was required.

In response to our letters, FSAR Supplement No. 22 concerning postulated high energy pipe ruptures outside containment was filed by PASNY on June 22, 1973, and supplemented by information in a letter submitted July 18, 1973.

B. EVALUATION

1. Criteria

A summary of the criteria and requirements is set forth below:

(a) Protection of equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from all effects resulting from ruptures in pipes carrying high-energy fluid, up to and including a double-ended rupture of such pipes, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig, respectively. Breaks should be assumed to occur in those locations specified in the "pipe whip criteria." The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement) and environmental.

(b) In addition, protection of equipment and structures necessary to shutdown the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying fluid routed in the vicinity of this equipment. The size of the cracks should be assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

2. High Energy Systems

Our evaluation included the following piping systems containing high energy fluids.

Main, Extraction and Auxiliary Steam Systems

Feedwater System

Condensate System

Reactor Core Isolation Cooling System

High Pressure Coolant Injection System

Reactor Water Cleanup System

Residual Heat Removal System

Sample Lines (Environmental Effects Only)

3. Targets of High Energy Pipe Breaks

The effects of high energy pipe breaks were evaluated on the following systems, components, and structures which would be

necessary (in various combinations, depending on the effects of the break) to safely shut down, cool down, and maintaining cold shutdown conditions:

(a) General

1. Control Room
2. Control and Instrument Cables and Tunnels
3. Electrical Distribution System
4. Emergency DC Power Supply (batteries)
5. Emergency AC Power Supply (diesels)
6. Heating and Ventilation Systems (needed for long term occupancy to maintain the reactor in safe shutdown condition.)

(b) Reactor Control Systems and associated instrumentation

(c) Cooling and Service Water Systems

(d) ECCS components

4. Specific Areas of Concern

The applicant has examined all potential safety-related high energy line break locations and evaluated break consequences. We have reviewed all of this information, including the following specific areas of concern, where the potential consequences might be severe or where specific corrective action would further assure safe cold shutdown of the plant.

(a) Compartment Pressurization

Large pipe breaks including the double ended rupture of the largest pipes in a system and small leakage cracks up to the design basis size have been considered for the areas housing ECCS equipment (crescent area), the main steam tunnel, and the turbine building.

In the main steam tunnel the effects of a combined main steamline break with that of feedwater line rupture were considered as a worst case. The resultant pressure was calculated to increase to 5.2 psi which is below the 6 psi design basis.

The largest energy producing break in the crescent area was assumed to be the break of the HPCI steamline to the turbine. The resultant energy release could cause a pressure buildup to 2.5 psi, which is below the design basis of 3 psi.

(b) Pipe Whip

Both the steam tunnel and crescent area have been designed with thick reinforced concrete capable of withstanding large static and dynamic loads. The reinforced concrete steam tunnel from the primary containment to the turbine room in which the main steam and feedwater lines are

routed is subjected only to the loads of the piping and a live load from the floor on top of the tunnel roof of 200 psf. The tunnel walls and roof have been evaluated for impact load of 500,000 lb caused by the postulated pipe break of the main steamlines and have been found to be satisfactory; that is, they will not break or spall, thus not creating any secondary missiles or exposing any safeguards systems to any damaging forces. The existing seismic restraint on the main steam lines will be modified to a whipping restraint to ensure that a postulated break in that area will not affect the Main Steam Isolation Valves.

The exterior walls of the crescent area housing ECCS equipment were designed to resist the earth pressures of the 50 ft of ground and the loads from the top slab. The interior wall was originally designed to carry the loads of the top slab of the crescent area and is also capable of carrying the pressure build-up due to a postulated pipe break in the crescent area within the code allowable stress. The top of the slab of the crescent area is designed to carry all equipment loads and a live load of 350 psf and is capable of carrying the pressure build-up in the crescent area due to the postulated pipe break.

(c) Control Room Habitability

The main control room is physically located away from and isolated from all high energy lines. Neither the control room equipment nor its ventilation system will be affected by environmental effects caused by a rupture of a high energy line.

(d) Torus Area

High energy lines to the HPCI and RCIC turbine inlet run above the torus, and penetration of the torus may occur if this line were to rupture. Modifications will be made to this area to prevent breaking of the torus by the addition of impact plates over areas where possible pipe whip could occur. Other high energy lines such as the sample lines and reactor water cleanup lines are located such that their rupture would not cause damage to the torus.

(e) Environmental Effects

Components and equipment were analyzed and physically checked for possible adverse environmental effects which could be caused by the rupture of a high energy line. Adverse temperature, pressure, and humidity were the parameters which were used in the evaluation of safety related equipment. We have reviewed the licensee assessment of the consequences of environmental effects on

safety equipment. We find that safety related equipment has been designed to limits in excess of postulated conditions which could arise from the rupture of a high energy line.

5. Modifications

Modifications to the existing facility are currently being undertaken by PASNY in order to assure that the design will have adequate safety margins in the event of a high energy line rupture outside the containment. The following is a summary of these modifications:

(a) Additional restraints and modifications to existing restraints will be added to the main steamline to ensure that a whipping steamline will not affect the Main Steam Isolation Valves.

(b) Impact plates will be provided for the steam supply to both the HPCI and RCIC turbines ensures the acceptability of the consequences of any postulated pipe break inside the torus compartment.

C. CONCLUSIONS

We have reviewed the information submitted to us and based on this review and our discussion with PASNY, find that the assessment of the consequences of high energy line failures outside containment is acceptable. Some modifications are necessary. The applicant has stated in his letter of June 13, 1974 that all modifications will be complete before initial fuel loading. We concur

with the applicant's evaluation that the potential consequences of these postulated high energy pipe failures, with the modifications, will not prevent the capability to achieve safe cold shutdown conditions consistent with the single failure and redundancy requirements as described in our letter of December 18, 1972, and thus there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner proposed.

10.7 Radioactive Materials Safety

The personnel qualifications, facilities, equipment and procedures for handling the byproduct, source and special nuclear material sources utilized for reactor startup and equipment calibration were reviewed. Based on the information provided in the FSAR, and amendments, we conclude that there is reasonable assurance that these sources will be stored and used in a manner to meet the applicable radiation protection provisions of 10 CFR 20 and 30.

12.0 QUALITY ASSURANCE

12.1 Independence of Quality Assurance

In a Commission Memorandum and Order dated December 7, 1973, concerning the LaSalle County Nuclear Station, Unit 1 and 2, the Regulatory staff was directed to determine, for facilities under construction and for construction applications under review, if quality assurance personnel have sufficient authority and organizational freedom to perform their critical functions effectively and without reservation. The staff subsequently sent PASNY a letter requiring this information for J. A. FitzPatrick.

We have reviewed and evaluated Power Authority of the State of New York (PASNY) letter of March 28, and referenced material, which is in response to our letter of February 22, 1974. The information provided by PASNY relates to the authority and organizational freedom of QA personnel within PASNY, General Electric, and Stone & Webster performing work on the design and construction of FitzPatrick Nuclear Power Plant.

This review and evaluation considered the duties, responsibilities, and authority vested in persons responsible for verification of quality by testing, inspecting, and auditing. We also reviewed the organizational arrangement and the administrative control and technical direction exercised over these persons with particular attention given to freedom and independence from the pressures of costs and schedules.

Based on our evaluation, we conclude that sufficient authority and organizational freedom existed to enable QA personnel to perform their critical functions effectively and without reservation for the design and construction of FitzPatrick Nuclear Power Plant.

Appendix A

Updated Chronology

February 1, 1973	Supplement No. 1 to the Commission's Safety Evaluation issued.
February 2, 1973	PASNY letter submits additional supporting information for previously submitted fracture toughness data.
February 2, 1973	PASNY letter provides additional information on vacuum breaker valve analysis.
March 1, 1973	PASNY letter revising the information contained in their letter dated 1-11-73 (Questions 2, 3 and 7 of AEC letter dated 12-21-72).
March 2, 1973	Special Prehearing Conference held.
March 5, 6, 7, 1973	Meeting with PASNY to discuss proposed Technical Specifications.
March 5, 1973	PASNY submits revisions to their letter of January 15, 1973.
March 16, 1973	AEC letter transmitting summary of meetings held on 3/5, 6 and 7/73.
March 19, 1973	Commission's Final Environmental Statement issued.
March 22 & 23, 1973	Meeting with PASNY to discuss proposed Technical Specifications.
March 22, 1973	AEC letter requests plan for providing spent fuel storage pool protection against all credible spent fuel cask drops.
March 28 & 29, 1973	Meeting with PASNY to continue discussions on proposed Technical Specifications.

April 5, 1973	PASNY letter in response to AEC letter of 3/22/73 indicating that a plan is being formulated.
April 12, 1973	PASNY letter providing a description for the main steam leakage collection system.
	PASNY letter submits description of main steam leakage collection system.
April 19, 1973	PASNY letter presents a probability study of missile damage from recirculation pump and motor overspeed.
April 26, 1973	Prehearing Conference held
May 4, 1973	AEC letter to PASNY presents review comments on main steamline leakage collection system.
May 4, 1973	PASNY letter providing justification for deletion of the heating system for the cooling water intake structure bars from their proposed Technical Specifications.
May 7, 1973	PASNY submits report on Effects of a High Energy Piping System Outside of Primary Containment.
May 7, 1973	Meeting with PASNY to discuss final changes in draft of proposed Technical Specifications.
May 10, 1973	AEC letter transmitting summary of the meeting held on May 7, 1973.
May 10, 1973	PASNY letter requests deletion of bar rack heater requirements from Technical Specifications.
May 11, 1973	ASLB's Notice of Hearing issued, which indicates that the environmental and safety hearings are to commence 6-4-73.
May 16, 1973	PASNY letter provides information on environmental qualification of safety/relief valve solenoids.

May 17, 1973	PASNY submits Amendment No. 21 together with Supplement No. 20 to the FSAR containing miscellaneous updated information.
May 18, 1973	Summary Statement by the Directorate of Licensing issued.
May 25, 1973	PASNY submits Amendment No. 22 together with Supplement No. 21 to the FSAR containing revised and updated Technical Specifications.
June 4 & 5, 1973	Public Hearing held in Oswego, New York.
June 7, 1973	PASNY letter advises that fuel loading date has slipped to January 1974.
June 8, 1973	PASNY letter commits to provisions which will reduce the probability of fuel cask drops into spent fuel storage pool.
June 21, 1973	PASNY submits Amendment No. 23 with Supplement No. 22 to FSAR.
June 25, 1973	PASNY letter advising that their schedule for the submittal of information on high energy and line breaker is July 22 and August 7.
June 28, 1973	PASNY submits Amendment No. 24 with Supplement No. 23 to FSAR.
June 29, 1973	PASNY letter transmits Stone & Webster report relating to the reactor pedestal crack.
July 17, 1973	PASNY submits Amendments 25 with Supplement No. 24 to FSAR.
July 18, 1973	PASNY submits additional information on high energy pipe ruptures.
July 19, 1973	AEC letter requesting information relating to "Model for Fuel Densification," which supplements the staff's letter dated 11/20/72.

July 20, 1973	PASNY letter transmitting the Environmental Technical Specifications.
July 27, 1973	PASNY letter providing a report describing their proposed main steam leakage collection system.
August 1, 1973	AEC letter granting the withholding of the security plan submitted with Amendment No. 24 pursuant to 10 CFR section 2.790(d).
August 13, 1973	Meeting held to discuss pedestal crack report.
August 17, 1973	PASNY submits request for CP extension.
August 20, 1973	PASNY submits latest information on fuel densification.
August 29, 1973	Received letter from PASNY changing fuel loading to April 1, 1974.
August 30, 1973	AEC letter transmitting summary of meeting held on 8-13-73.
September 10, 1973	PASNY submits supplemental information for request CP extension.
September 20, 1973	PASNY letter submitting material relating to the pedestal crack, which is a followup to meeting of 8-13-73.
September 24, 1973	AEC letter requesting additional information in connection with PASNY's 9-10-73 letter.
September 25, 1973	PASNY letter providing information requested by AEC's letter dated 9-24-73.
September 25, 1973	PASNY letter submitting details on the vibration testing program, in reference to Supplement 16, Question 3.8.
September 26, 1973	AEC letter requesting additional information in connection with the main steam leakage collection system.

October 9, 1973	CP was extended until July 1, 1974. (Order transmitted by AEC letter dated 9-28-73.)
October 10, 1973	Letter A. Giambusso transmitting the staff's evaluation of ATWS data.
October 19, 1973	AEC letter requesting that a positive repair method for the cracked pedestal be submitted.
October 26, 1973	PASNY submits suggested method of repair of cracked reactor pedestal.
November 1, 1973	ASLB issues Memorandum and Order on certain evidence from the Nine Mile Point 2 CP hearing.
November 12, 1973	The Initial Decision was rendered by the ASLB authorizing issuance of an operating license.
November 15, 1973	PASNY submits additional information on repair of cracked pedestal.
December 3, 1973	PASNY submits final drawings of pedestal repair method.
December 4, 1973	PASNY letter submitting additional information on the high energy fluid pipe breaks and the main steam leakage collection system.
December 7, 1973	AEC letter finding the repair method on the pedestal acceptable.
December 7, 1973	PASNY letter submitting information based on additional fuel densification calculations.
December 10, 1973	PASNY files a brief in support of exceptions to initial decision.
December 10, 1973	ASLB's Order amends the proposed operating license attached to the Initial Decision dated 11-12-73.
December 11, 1973	AEC letter transmitting a Notice of Availability of the Initial Decision... (published in the Federal Register 12/17/73; 38 FR 34684)

December 13, 1973	PASNY letter transmitting "Program for Requalification of AEC Licensed Personnel".
December 14, 1973	AEC letter granting the withholding of vibration test information submitted by letter dated 9/25/73.
December 17, 1973	PASNY letter discussing further information on the reactor internals (cold flow) vibration testing.
December 26, 1973	AEC letter specifying QA measures which should be approved by 1/31/74.
December 28, 1973	Appeal Board's Memorandum and Order which remands the proceeding to the Licensing Board for further consideration of the condition challenged by the applicant's exception.
December 28, 1973	PASNY letter responding to AEC 's 10/10/73 letter relating to ATWS.
January 10, 1974	ASLB's Supplemental Initial Decision issued.
January 24, 1974	PASNY letter responding to AEC's 12/26/73 letter relating to QA.
January 28, 1974	Meeting with PASNY to finalize the Technical Specifications.
January 29, 1974	Appeal Board's Decision, which affirms the ASLB's Initial Decision.
February 13, 1974	PASNY letter advising that the fuel loading date has been extended to 6/1/74.
February 20, 1974	AEC letter transmitting summary of meeting held on 1/28/74.
February 22, 1974	AEC letter requesting additional information related to radioactive materials.
February 22, 1974	AEC letter requesting information related to QA personnel, in connection with a Commission Memorandum and Order for LaSalle.

March 1, 1974	Meeting held for the purpose of discussing QA program for operation of FitzPatrick.
March 15, 1974	PASNY letter submitting radioactive materials information requested by AEC letter dated 2/22/74.
March 18, 1974	PASNY letter confirming information provided verbally on scram reactivity for reload cores.
March 18, 1974	AEC letter transmitting Notice of Availability of the Appeal Board's Decision, dated 1/29/74 (published 3/25/74; 39 FR 11131).
March 20, 1974	AEC letter transmitting summary of meeting held on 3/1/74.
March 28, 1974	PASNY letter submitting additional QA information.
April 2, 1974	PASNY letter transmitting a description of the Group Notch Rod Sequence Control System which is intended to be incorporated in the FitzPatrick Plant.
April 5, 1974	PASNY letter advising that a QA program has been submitted to RO for review.
April 22, 1974	AEC letter requesting that a physical security plan be submitted in accordance with Regulatory Guide 1.17.
May 3, 1974	PASNY letter submitting information related to the group notch rod sequence control system.
May 3, 1974	PASNY letter advising that the fuel loading date will be extended from 6/1/74 to July 15, 1974.
May 13, 1974	PASNY letter submitting additonal information related to the group notch rod sequence contrl system.
May 16, 1974	PASNY letter responding to AEC's 4/22/74 letter regarding the physical security plan.

May 20, 1974	PASNY letter providing information pertaining to the integrated leak rate test.
June 5, 1974	PASNY letter supplementing their 5/20/74 letter regarding the integrated leak rate.
June 13, 1974	PASNY letter requesting AEC approval of the main steam leakage collection system.
June 18, 1974	GE letter submitting affidavits to support statements being made by various utilities in their requests related to ECCS.
June 19, 1974	AEC letter requesting additional information related to the main steam leakage collection system.
June 19, 1974	AEC letter advising that in order for the physical security plan to be acceptable it must provide for armed guards.
June 20, 1974	PASNY letter requesting an exemption from the requirements of Appendix K of 10 CFR Part 50 until 10/1/75 or the first major refueling outage, whichever is later.
June 20, 1974	PASNY letter requesting an extension until 10/4/74 for submitting ECCS evaluations.
June 28, 1974	PASNY letter discussing problems with the ultrasonic baseline inspection.
June 28, 1974	PASNY letter withdrawing its letters dated 5/20 and 6/5/74 relating to integrated leak rate testing.
July 1, 1974	PASNY submits a request for an extension of CPPR-71.
July 3, 1974	PASNY letter advising that the fuel loading date has been extended from July 15, 1974 to October 1, 1974.
July 9, 1974	AEC letter transmitting Notice of Request for Extension in Submitting Evaluations (published 7/10/74; 39 FR 25417).

July 10, 1974	Notice of Request for Exemption from Requirements Concerning ECCS Performance published in the Federal Register (39 FR 25420).
July 10, 1974	PASNY letter providing information pertaining to MAPLHGR curves as related to GEGAP III described in NEDO 20181, Rev. 1; and NEDC 20181, Rev. 1 (Class III).
July 18, 1974	PASNY letter responding to AEC's 6/19/74 letter regarding the main steam leakage collection system.
July 18, 1974	PASNY letter providing information related to armed guards.
July 31, 1974	AEC letter transmitting "Comments of the Director of Regulation on Requests for Exemptions from the Requirements of 10 CFR Section 50.46".
August 5, 1974	The Commission's "Memorandum and Order" related to requests for exemption from ECCS issued.
August 6, 1974	AEC letter transmitting the "Determination of Requests for Extension in Submitting Evaluations".
August 12, 1974	AEC letter requesting additional information related to PASNY's 7/1/74 request for extension.
August 14, 1974	PASNY letter responding to AEC's letter dated 8/12/74.
August 29, 1974	Amendment No. 26 (Supplement 25) submitted by PASNY letter.
August 30, 1974	PASNY submitted Draft of Appendix K material.
September 12, 1974	AEC letter requesting further information on the Industrial Security Plan.
September 17&18, 1974	Meeting held at the site between AEC and PASNY representatives to discuss outstanding items of plant construction; final Technical Specifications; and the operating agreement between PASNY and Niagara Mohawk Power Corporation.