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New York Power Authority

William J. Cahill, Jr.
Chief Nuclear Officer

November 20, 1996
JPN-96-045

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: **James A. FitzPatrick Nuclear Power Plant**
Docket No. 50-333
Incorrect Attachments Included with Response to Request for Additional Information on Power Uprate

Reference: NYPA letter, W. J. Cahill, Jr. to USNRC dated November 14, 1996 (JPN 96-043) regarding response to request for additional information regarding power uprate.

Dear Sir:

Due to an administrative error, the wrong attachments were included with the Reference letter. Attached are the correct Attachments. These supersede and replace those attachments included with the Reference letter.

We regret any inconvenience this error may have caused. If you have any questions, please contact Ms. C. Faison.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Chief Nuclear Officer

cc: see next page

A001 1/1

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Attachments

1. Attachment I to JPN-96-043, Request for Additional Information Regarding Power Uprate
2. Attachment II to JPN-96-043, Requested Changes to the Facility Operating License
3. Attachment III to JPN-96-043, Technical Specification Page 254c Changes
4. Attachment IV to JPN-96-043, Mark-up of Technical Specification Page 254c
5. Attachment V to JPN-96-043, Summary of Commitments

Attachment I to JPN-96-043

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 1

Provide evaluation of the effect caused by reactor uprate on erosion/corrosion of components exposed to single and two phase fluids. The proposed increase of reactor power would cause a corresponding increase in the operating temperatures, pressures, and flows in the BOP systems. These parameters would have a significant effect on the rates at which the components susceptible to erosion/corrosion are degraded. In order to ensure safe operation of the plant provide erosion/corrosion analyses.

Response to Question 1

The potential affects of increased pressure, temperatures, and flowrates on the Flow Accelerated Corrosion (FAC) models were reviewed (Reference 1) as part of the power uprate program. These analyses conclude that, based on the relatively small increase in these parameters under uprate conditions, there will be a negligible effect on Erosion/Corrosion (E/C) wear rates. As a result, their relative ranking for inspection priority is not expected to change.

The negligible increase in E/C rates will not affect any end-of-life predictions for components previously inspected.

The Authority uses EPRI's CHECMATE/CHECWORKS Flow Accelerated Corrosion (FAC) model as part of the FAC monitoring program at FitzPatrick. Highly susceptible FAC points will be inspected based on CHECMATE/CHECWORKS analyses during the current Refuel 12/Cycle 13 (R12/C13) Refueling Outage (RFO), which started October 26, 1996. These inspection points were selected using the pre-uprate parameters.

Future Inspections

The FAC models will be updated and computer analyses will be performed to include the inspection data from R12/C13 and the power uprate heat balance inputs. These updates are required as part of plant procedures and will be performed prior to the R13/C14 RFO. Inspection points for the next RFO (i.e., R13/C14, which is currently scheduled to commence in the Fall 1998) will be based on these new analyses.

Depending on the extraction line selected, extraction steam flow rates will increase by approximately 1.0% to 12%. The overall effect of the increased extraction steam flow is not expected to alter the E/C rates significantly because increases in velocity, when adjusted for changes in fluid specific volume, are minimal.

The condensate and feedwater system will have uprate flows approximately 5% higher than the current normal operating conditions. Reference 1 concluded that this will result in a minor increase in E/C rates.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 2

Is there assurance that a similar excessive recirculation pump vibration problem will not occur at JAFNPP for power uprate conditions that occurred at Susquehanna?

Response to Question 2

Power uprate analyses for FitzPatrick does not require an increase in core flow. The Authority expects only a slight increase (approximately 2%) in recirculation pump speed to achieve rated core flow. Therefore, a negligible effect on pump vibration is expected.

FitzPatrick will monitor recirculation pump vibration during startup from the R12/C13 RFO utilizing existing monitoring equipment installed for this purpose.

The Authority does not expect a vibration problem at FitzPatrick similar to that which occurred at Susquehanna, as described in NRC Information Notice 95-16, "Vibration Caused by Increased Recirculation Flow in a Boiling Water Reactor," dated March 9, 1995.

As requested, a license condition requiring recirculation pump motor vibration monitoring during initial power ascension to uprated power conditions will be added to the facility Operating License (OL) as described in Attachment II.

Question 3

Does the offsite and the onsite power distribution systems/grid stability conform with GDC17, Appendix A to 10 CFR Part 50?

Response to Question 3

The plant electrical systems were evaluated by the Authority to confirm that these systems will perform their intended functions following power uprate. The results of this evaluation confirm that uprate can be accommodated. This was described in NEDC-32016P-1 Section 6.1 and 6.2 (Reference 2).

As part of a power uprate study, which began in 1995, the Authority identified a potential transient stability issue on the 345 kilovolt system outside of the plant. This condition existed under the current operating conditions (i.e., pre-power uprate). Based on this, terminal voltage operating guidelines were put in place to ensure the stability of the unit. Revised terminal voltage operating guidelines are to be implemented to reflect the power uprate following approval of the Authority's power uprate submittal.

Stability simulations were conducted with the FitzPatrick unit loaded to 883 MW gross electrical output. Stability swing curves (Curves 1A and 1B attached) show well damped and stable results for a loss of FitzPatrick generation. Stability swing curves (Curves 2A and 2B attached) show well damped and stable results for the most severe normal criteria contingency (i.e., stuck R935 breaker at Scriba).

Based on the revised operating guidelines that will be in place, the FitzPatrick unit will remain stable under power uprate conditions.

Updated Final Safety Analysis Report (UFSAR) Section 16.6, titled "Conformance to AEC Design Criteria", states the following, in part:

"The James A. FitzPatrick Nuclear Power Plant was evaluated against the USAEC Design Criteria, 10 CFR 50 Appendix A, General Design Criteria for Nuclear Power Plants, effective May 21, 1971.

The design bases of the James A. FitzPatrick Nuclear Power Plant were evaluated against each of the six groups of the criteria. In each group the interpretation of the intent of the criteria is stated, and the plant design conformance to this interpretation is discussed."

Response to Question 3 cont'd

"Based on the understanding of the intent of these criteria, it was concluded that the James A. FitzPatrick Nuclear Power Plant conformed with the intent of the AEC General Design Criteria for Nuclear Power Plants to the maximum extent possible consistent with the state of design and construction at the time of issuance of these criteria."

"16.6.2.2 Group II: Protection by Multiple Fission Product Barriers (Criteria 10-19)

Interpretation and Conclusion. The criteria in Group II are intended to assure that designs have provided the reactor plant with multiple barriers against the release of fission products to the environs and that these barriers remain intact during normal operations and all operational transients caused by a single operator error or equipment malfunction, and that adequate barriers are available for design basis accidents. In addition, these criteria are intended to identify and define the instrumentation and control systems, electrical power systems, and control room requirements necessary to maintain the plant in a safe operational status.

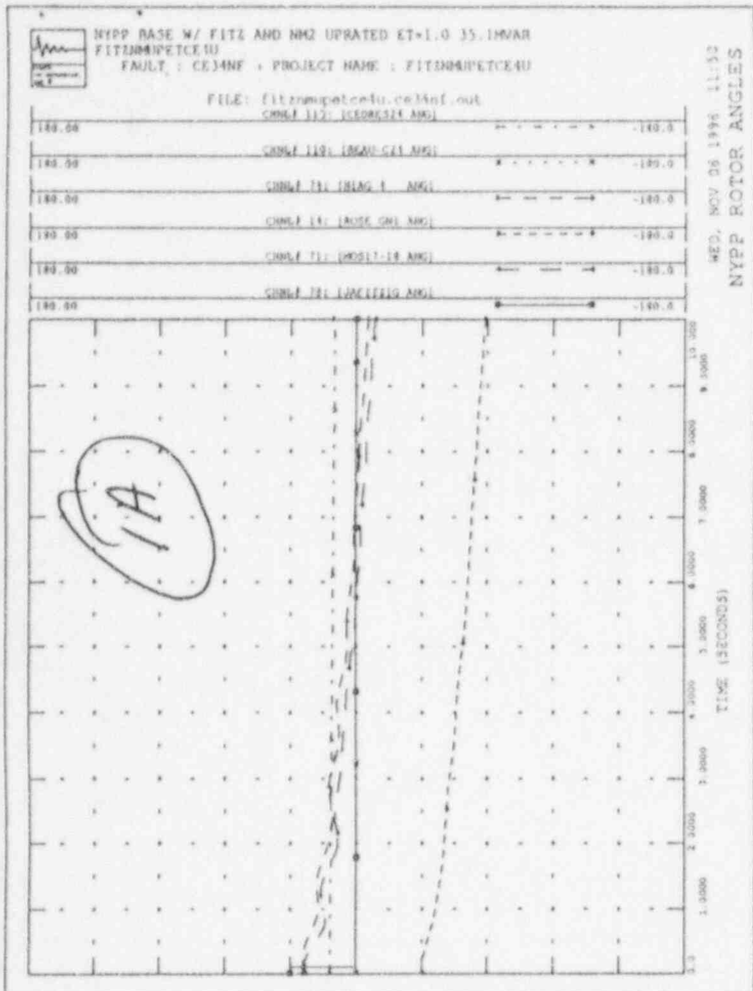
It is concluded that the design of this plant conforms with the criteria of Group II based on the above interpretation of the intent of these criteria."

"The capacity and capability of the onsite and offsite electrical power systems are adequate to accomplish all required safety functions independently under postulated design basis accident conditions (Criterion 17)."

Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission, in the Matter of Power Authority of the State of New York, James A. FitzPatrick Nuclear Power Plant Docket No. 50-333, issued November 20, 1972 states the following, in part, regarding conformance with General Design Criteria:

"Based on our evaluation of the design and design criteria for the James A. FitzPatrick Nuclear Power Plant, we conclude that there is reasonable assurance that the intent of the General Design Criteria for Nuclear Power Plants, published in the Federal Register on May 21, 1971 as Appendix A to 10 CFR 50, will be met."

The Authority has evaluated the effect of the power uprate on the necessary electrical systems and components. The results of these evaluations show that the safety functions of the electrical power system will be maintained under power uprate conditions. Based on this, the Authority concludes that the design of the FitzPatrick plant meets the intent of Criteria 17 as described in the SAR.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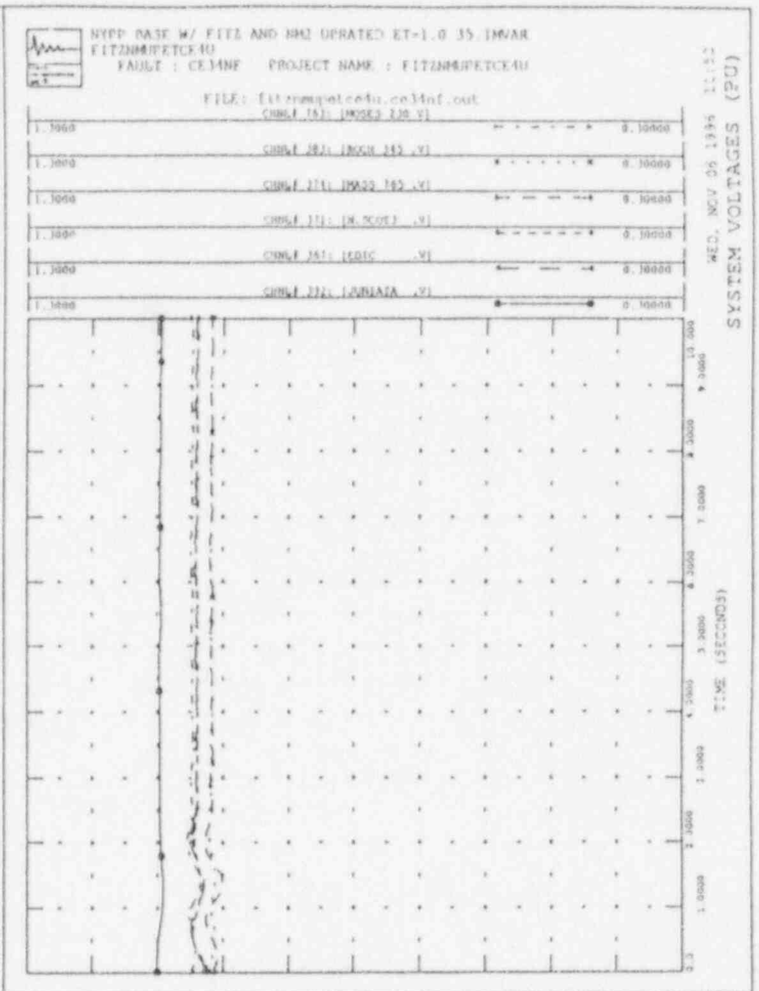
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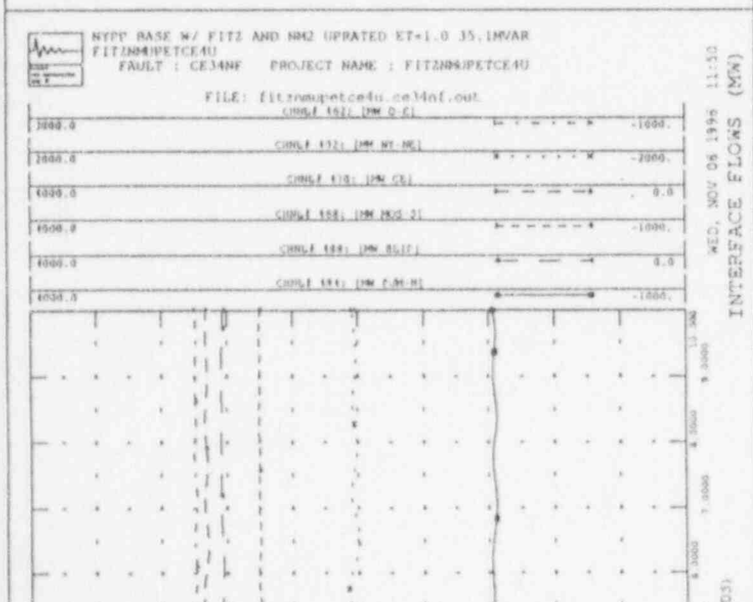
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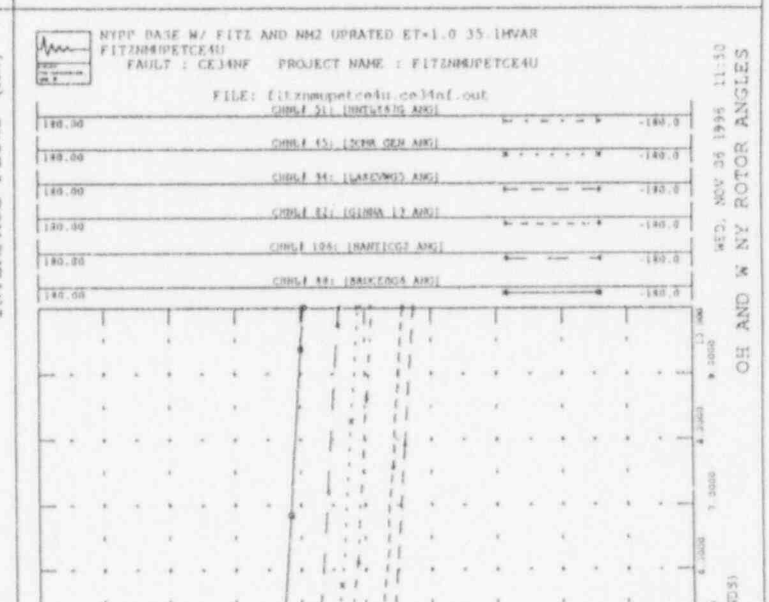
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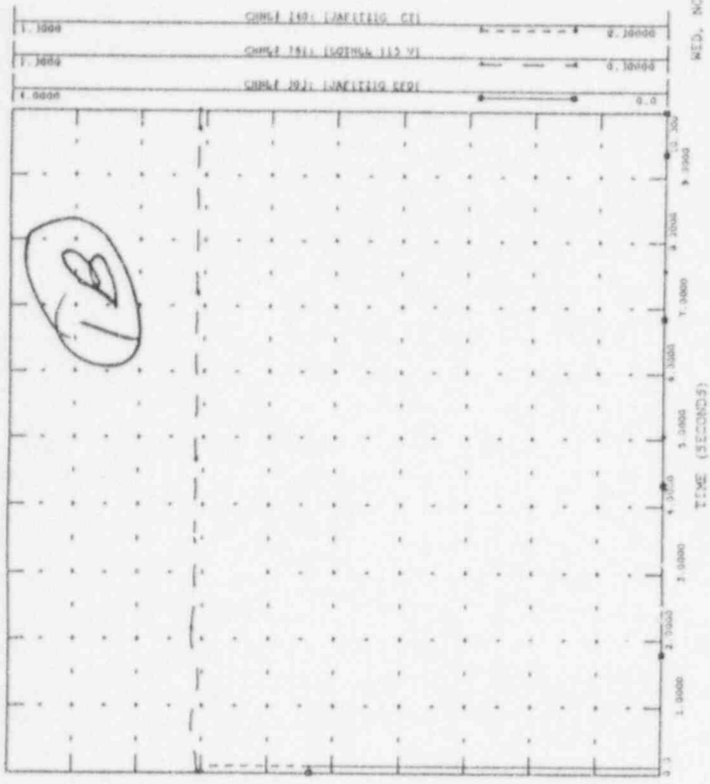

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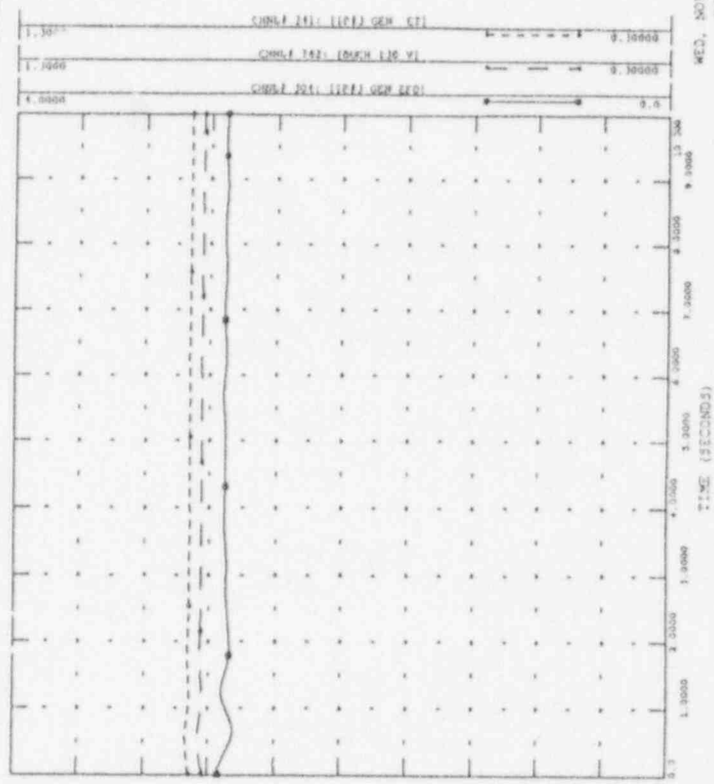

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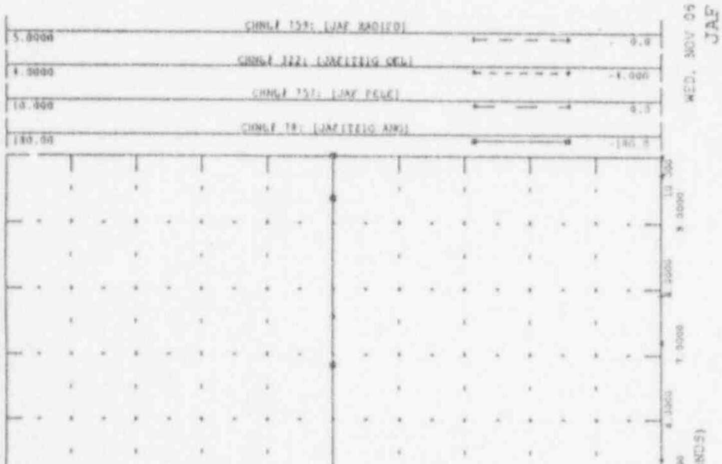

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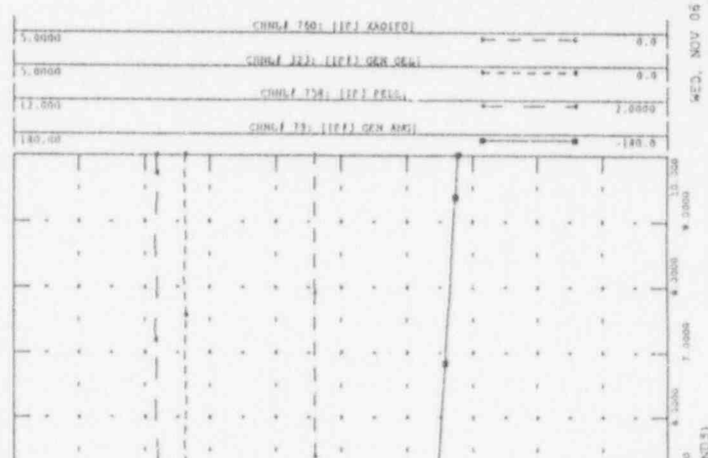

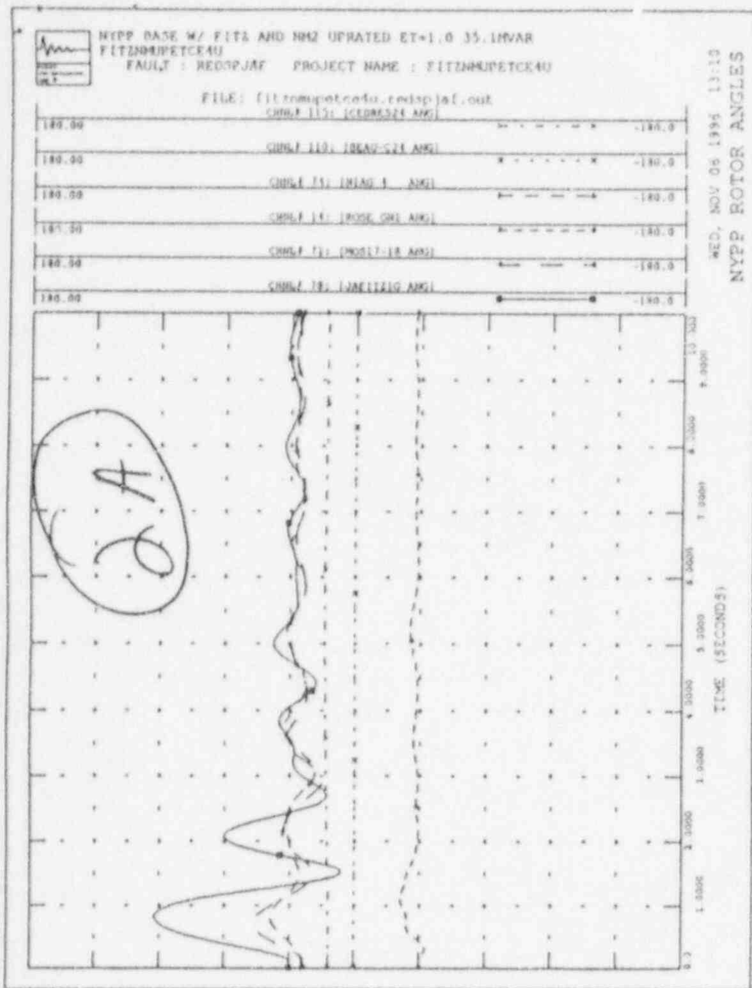
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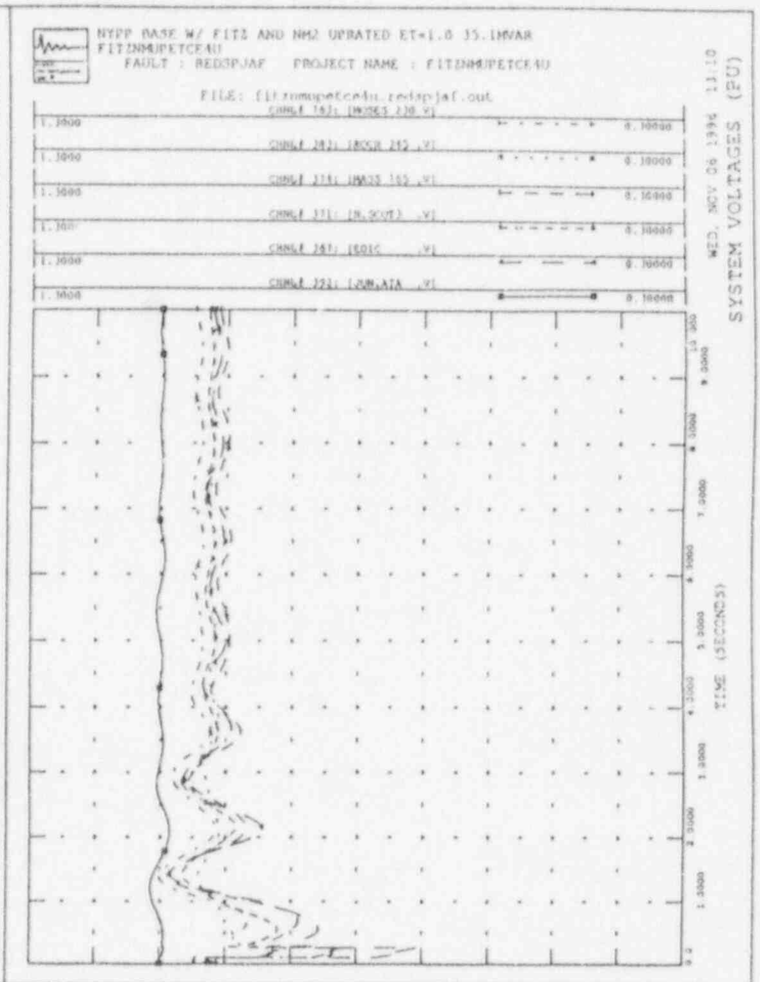
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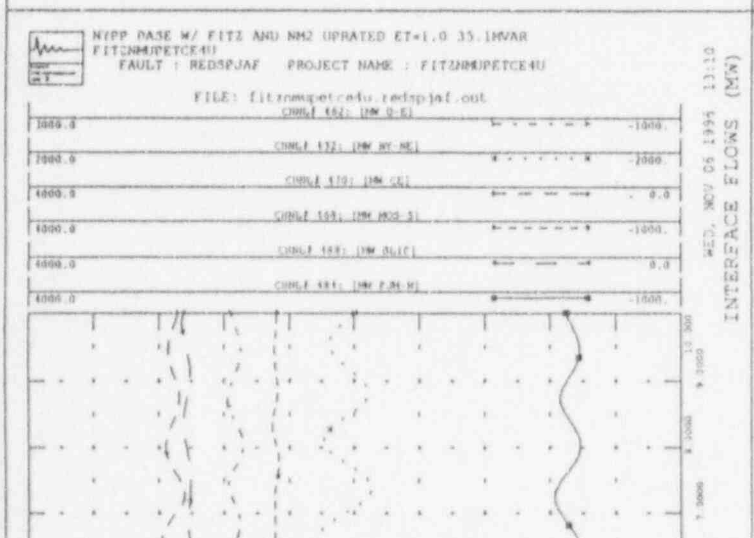
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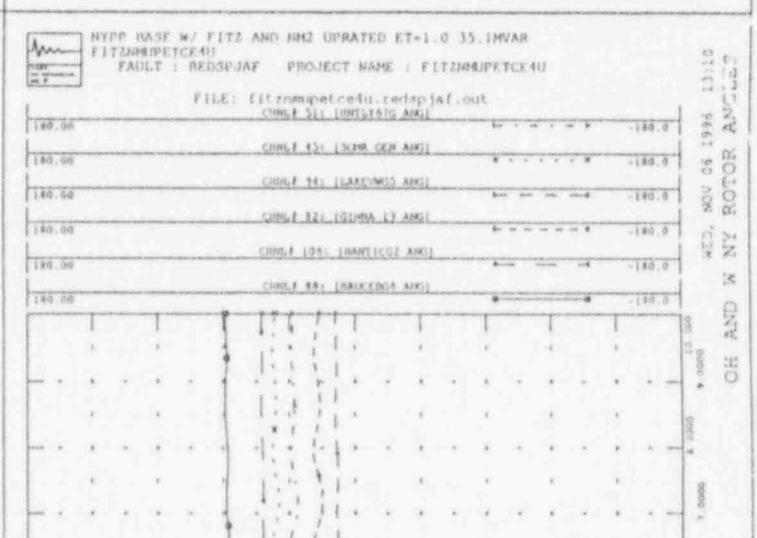
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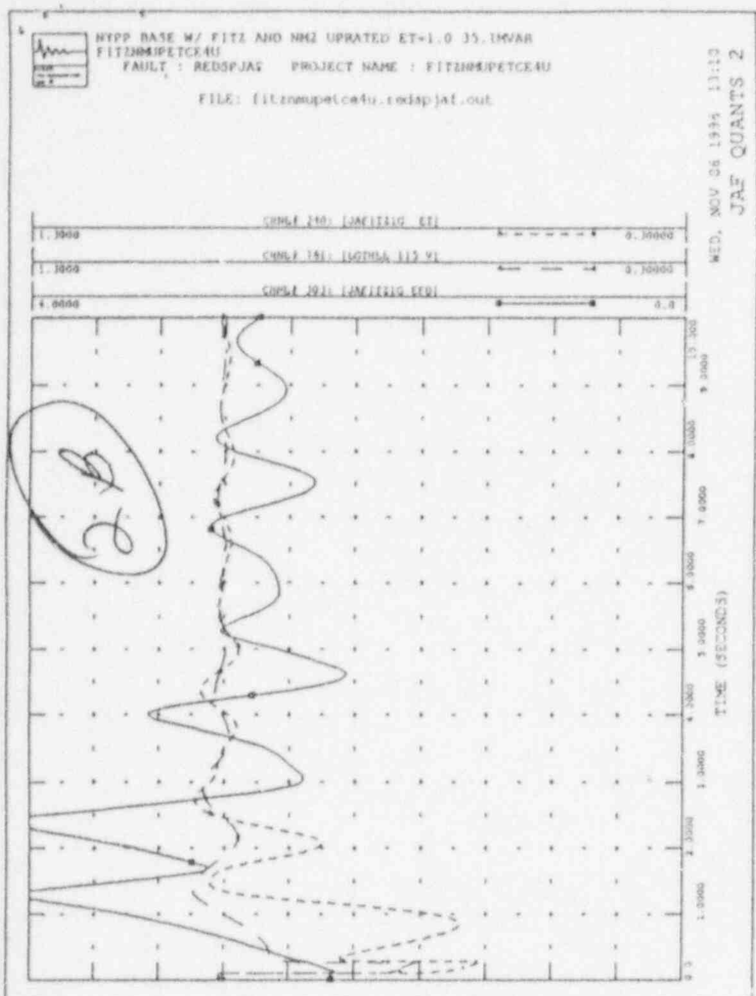
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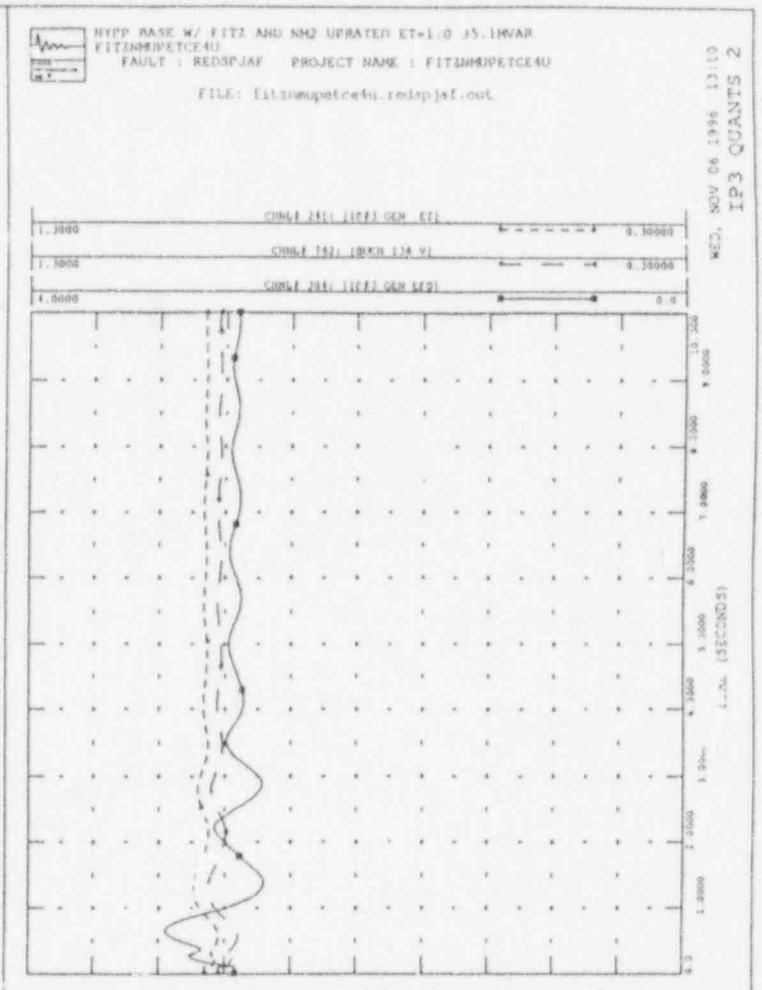
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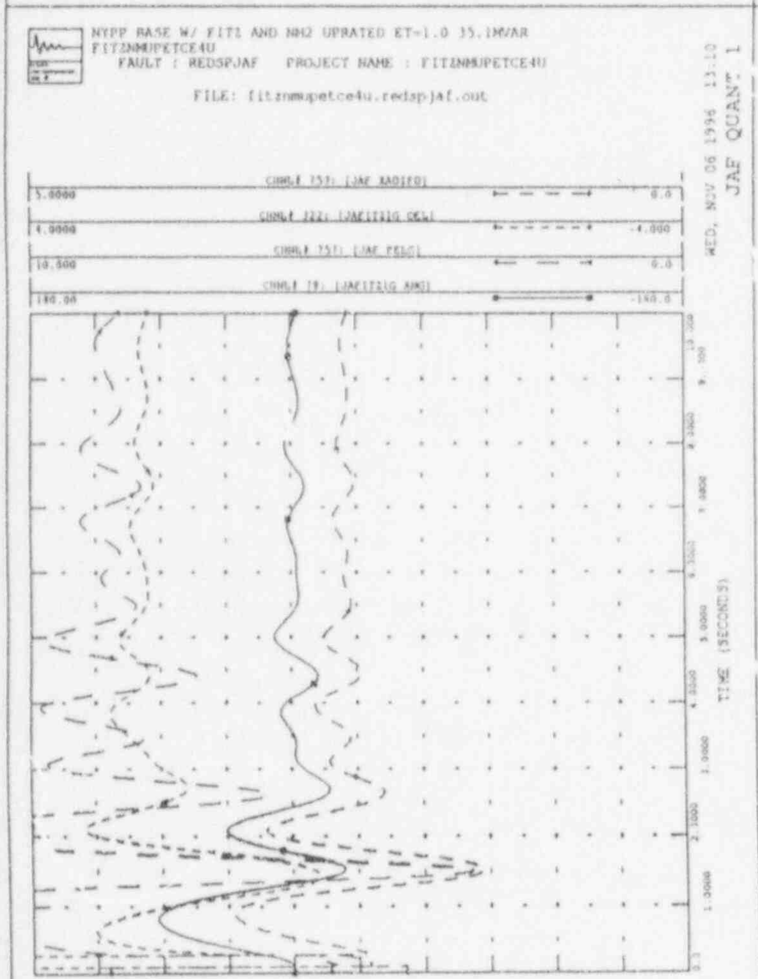
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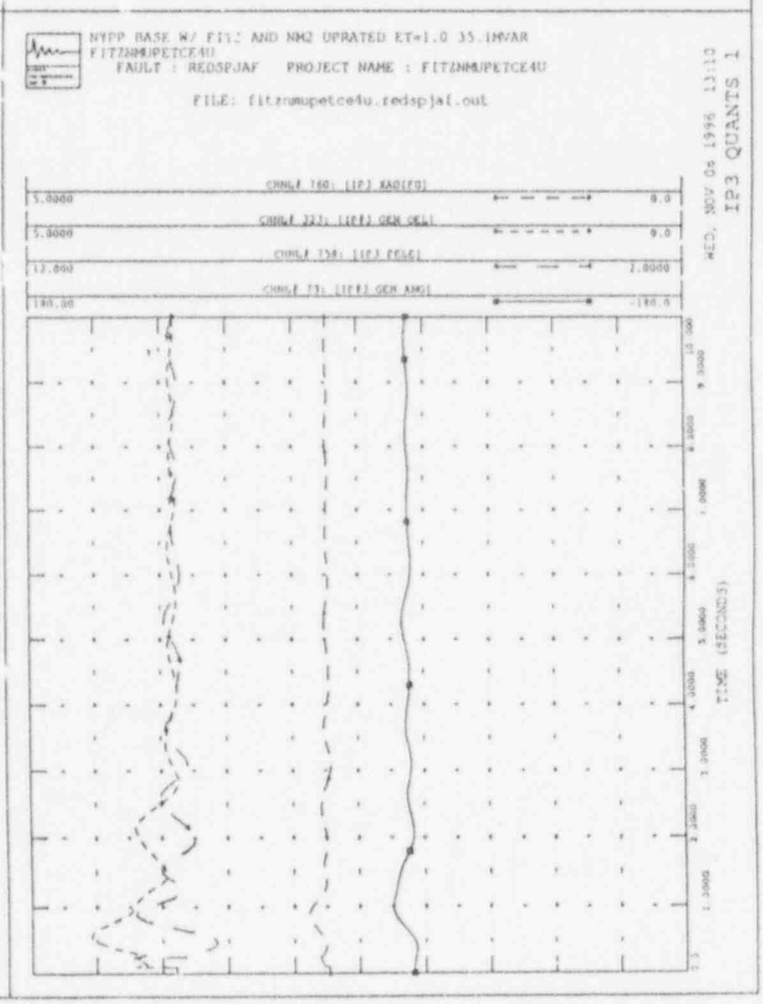
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TIME (SECONDS)

Question 4

Define references 6.9(A)4.b.1 and 6.9(A)4.b.2 on page 254c of the Technical Specifications.

Response to Question 4

The revision of NEDE-24011-P which provides the analytical methods supporting the Reload 12 / Cycle 13 core is "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13, August 1996. The revision of NEDC-31317P which supports the Reload 12 / Cycle 13 core is "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," NEDC-31317P, Revision 2, April 1993.

Attachment III provides a revised Technical Specification page 254c which includes the specific revisions of NEDE-24011-P-A and NEDC-31317P used as a basis for the Reload 12 / Cycle 13 reactor core. Item 6.9(A)4.b.3 "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda is deleted since it no longer supports the reload core. Items 6.9(A)4.b.4 and 6.9(A)4.b.5 added in amendment 236 have been renumbered accordingly. The specified revisions of NEDO-31960-A and NEDO-31960-A, Supplement 1 are those which support the Reload 12 / Cycle 13 core.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 5

What is the effect of delta P across MOV's?

Response to Question 5

The Authority evaluated the effect of power uprate Line Pressures (LPs), Differential Pressures (DPs), and post-accident maximum ambient temperatures on Motor Operated Valves (MOVs) within the scope of GL 89-10. As a result of these evaluations, no MOV modifications or field adjustments (i.e., torque or limit switch adjustments) will be required to support power uprate.

The FitzPatrick MOV group was provided with power uprate values for the following:

- Maximum normal operating pressure
- Safety Relief Valve (SRV) setpoint
- Post-LOCA Maximum drywell pressure
- Post-LOCA Maximum wetwell pressure

The scope of GL 89-10 MOVs that had design basis DPs and LPs affected by the power uprate values for the parameters noted above were then determined and new power uprate, DP, and LP evaluations were performed. Power uprate MOV design calculations were then performed utilizing the new power uprate, DP, and LP values that resulted from the evaluations. As a result of these calculations, the Authority concluded that no MOV modifications or field adjustments (i.e., torque or limit switch adjustments) are required to support power uprate.

In November 1995, Engineering concluded that post-accident maximum ambient temperatures would not increase by more than 5°F, anywhere in the plant, as a result of power uprate. Therefore, in the absence of specific values, the FitzPatrick MOV group conservatively determined the effect of a 10°F increase in post-accident maximum ambient temperatures on each MOV within the scope of GL 89-10. This resulted in a torque derate which was then compared to each MOV's minimum thrust/torque requirements. As a result of this evaluation, the Authority concluded that no MOV modifications are required to support power uprate.

Question 6

Does power uprate have any effect on equipment qualification of mechanical equipment and non-metallic components (increased radiation)?

Response to Question 6

The affect of power uprate on environmental qualification has been evaluated. The changes in operating temperature, pressure, flow, and radiation due to power uprate will not adversely affect the qualification of mechanical equipment. The Authority evaluated these parameters and determined that the increases are enveloped by the original design of the plant equipment.

With respect to radiation, General Electric determined that the calculated doses resulting from power uprate are lower than the calculated doses prior to power uprate. The reason for this is the calculated radiation source terms for uprate, based on newer computer codes, are actually lower than the original calculated source terms.

The accident radiation doses used at FitzPatrick for environmental qualification prior to power uprate were calculated by Stone and Webster. The Authority recalculated the accident radiation doses to reflect power uprate and to reflect changes in airborne radiation doses. The net result from these two factors was an increased accident dose in some plant locations and a decreased accident dose in other plant locations.

For mechanical equipment, the materials of concern are typically elastomeric seals, O-rings, and other non-metallic pressure retaining parts. Testing of elastomers summarized in EPRI NP-1558 (Reference 3) shows that mild to moderate damage occurs at doses greater than approximately 5×10^5 rads. None of the revised accident doses adversely affect these materials.

The largest percentage increases in calculated accident radiation dose are in plant locations where the original accident radiation dose was less than 10^5 rads. The increased accident radiation doses for these plant locations, due to the revised contribution of airborne radiation, remain below 10^5 rads. For plant locations where the original accident radiation dose is greater than 10^5 rads, the largest percentage increase in radiation dose was from 2.3×10^5 rads to 3.49×10^5 rads. For plant locations where the original accident dose was greater than 5.0×10^5 rads, the largest percentage increase was from 4.5×10^6 rads to 4.8×10^6 rads.

Based on the above, the Authority concludes that the power uprate will not adversely affect mechanical equipment and, therefore, will not affect the ability of the mechanical equipment to perform its intended function.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 7

Describe the power uprate startup testing program.

Response to Question 7

Plant specific safety analysis for FitzPatrick (NEDC-32016P-1, Reference 2) states that a testing plan will include the testing requirements outlined in Section 5.11.9 and Appendix L.2 of NEDC-31897P-1 (Reference 4). These are the only sections that pertain to testing.

The Authority has analyzed the requirements in NEDC-32016P-1 and NEDC-31897P-1 and compared them with original startup testing. As stated in NEDC-31897P-1, the test plan will include tests for systems or components which have revised performance requirements.

JAF-RPT-MULTI-02420 (Reference 5) lists which systems and components will need to be tested based on revised performance requirements, and acceptance criteria from original startup testing. Reference 5 also lists the conditions and the order for the tests. The majority of the testing will be done utilizing existing or revised procedures. New tests have been or are being written for Reactor Core Isolation Cooling (RCIC), Feedwater, Electro-Hydraulic Control (EHC) and, Steady State Data.

Turbine control valve position as a function of reactor power (i.e., to 100% uprated power) will be one of the operational checks made of the pressure control system. Additionally, step changes (i.e., up to 10 psi) will be introduced into the in-service pressure regulator (both pressure regulators will be tested) to observe system response. This test will last be performed at 96% uprated power. The FitzPatrick initial startup test program included the same test at 96% of initial licensed power.

As requested, a license condition requiring performance of a power uprate startup test program will be added to the facility Operating License (OL) as described in Attachment II.

Question 8

Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will it require any new operator actions?

Response to Question 8

Power uprate does not change the type or scope of plant emergency and abnormal operating procedures. Power uprate does not change the type, scope or nature of operator actions required for accident mitigation and adds no new actions.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 9

Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with a power uprate. Please specify the expected response times before the power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to power uprate? Please state why reduced operator response times are needed. Please state whether reduced time available to the operator due to the power uprate will significantly affect the operator's ability to complete manual actions in the times required.

Response to Question 9

- Examples of operator actions potentially sensitive to power uprate are: initiation of emergency RPV depressurization in response to a loss of high pressure injection systems and initiation of the Standby Liquid Control system (SLC) following a high-powered Anticipated Transients Without Scram (ATWS).

- Power uprate affects the time available to the operator to perform actions primarily through the increase in power (reflected in decay heat rate following a scram and the heat which must be rejected from or accommodated within containment for ATWS events). As such, the time available to take action is generally inversely proportional to the power (there are secondary non-linear affects such as changed thermodynamic properties of water in the RPV, etc.). No significant effect on the operators' ability to complete actions in response to normal operation, transients or accidents has been observed during simulator training utilizing simulated power uprate conditions compared to similar observations for the current rated power. The simulated time for RPV water level to lower from normal operating level to top of active fuel on a loss of high pressure injection systems was approximately 3% shorter for an initial power level of 2536 MW_t than for an initial power of 2436 MW_t. For an ATWS event with MSIV closure, the simulated time for a torus water temperature increase from 79°F to 110°F (Boron Injection Initiation Temperature) changed from 101 seconds at an initial pre-transient power level of 2436 MW_t to 96 seconds for a pre-transient power level of 2536 MW_t.

- In general, operator reliability depends upon, among other factors, the time available for diagnosing and executing a particular action. Power uprate could decrease the time available for the operator action and therefore impact the human reliability analysis (HRA) and the probabilistic risk assessment (PRA) results. However, this is only true for actions which require diagnosis within a relatively short time frame (i.e., less than 30 minutes). As the time available increases, the associated diagnosis human error probability (HEP) becomes smaller and the overall HEP becomes dominated by non-time dependent human errors (i.e., errors of omission and commission). In addition, even in cases where the human error probability is doubled, the resulting impact on the total

Response to Question 9 cont'd

core damage frequency needs to be considered as well. As a rule, the percentage increase in core damage frequency is much smaller than the percentage increase in any one error probability.

The following table provides examples of operator actions that are potentially sensitive to power uprate and shows the impact of power uprate on the resulting HEPs. It should also be noted that the current mean HEPs correspond to the updated Individual Plant Examination (IPE) HRA, and the mean HEP with the power uprate reflects changes in the updated HEPs.

Response to Question 9 cont'd

Operator Action	Current Mean HEP	Mean HEP w/ Power Uprate	Percent HEP Change	Resultant Point Estimate (CDF)	Percent CDF Change
Failure to override MSIV isolation during an ATWS	0.42	0.54	28.6%	1.55E-6	<0.1%
Failure to initiate SLC during an ATWS w/ MSIVs closed	5.0E-3	5.6E-3	12.0%	1.55E-6	<0.1%
Failure to maintain level at top of active fuel during an ATWS w/ MSIVs closed	0.22	0.25	13.6%	1.55E-6	<0.1%
Failure to defeat HPCI auto transfer on high torus level during an ATWS w/ MSIVs closed	0.079	0.092	16.5%	1.55E-6	<0.1%
Failure to depressurize RPV during an intermediate LOCA w/ loss of high pressure injection	9.4E-3	1.0E-2	6.4%	1.55E-6	<0.1%
Failure to align RHRSW for injection during a small LOCA	0.030	0.032	6.7%	1.55E-6	<0.1%
Failure to manually open LPCI injection valves	0.29	0.32	10.3%	1.56E-6	0.65%

Total %CDF Change = <1.0%

In calculating the HEPs associated with power uprate, the time available for diagnosis was conservatively reduced by 5% and a new diagnosis HEP was calculated. The diagnosis HEP was then added to the post-diagnosis (non-time dependent) HEP to derive a value of the overall mean HEP for a particular action.

Response to Question 9 cont'd

All of the above changes in estimated HEPs are within the range of uncertainty for those probabilities. Furthermore, the four operator actions associated with ATWS events which result in the highest increase in human error probability have low risk significance in terms of their impact on the total core damage frequency. Therefore, the increase in the overall core damage frequency attributed to power uprate related increases in human error probabilities are within the range of uncertainty of the overall core damage frequency and are probabilistically insignificant.

Simulator training scenarios which incorporated these expected responses were included in approximately 10 hours of simulator training during cycle 96-6, which used a simulator model based on power uprate system parameters. Review of the crew self-assessments, (there are 6 operating crews and 4 staff crews), revealed there were no operational or simulator problems identified.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 10

What are the changes, based on power uprate, to control room instruments, alarms, and displays? Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)?

Response to Question 10

There are seven new meters on the control room 09-5 panel which are used for turbine control. These replacement meters are very similar to the old meters, but have higher upper limits. These new meters are needed since the new operating points are beyond the upper limit of the existing meter scales.

There is a recorder for main steam flow that requires a new scale. This is required since the upper limit for this recorder would not accommodate the new steam flow under power uprate conditions.

There are several scales in the control room that have normal operating bands or trip setpoint indications which have been adjusted for the new normal RPV pressure and high RPV pressure trip setpoints.

All instrumentation changes have been prepared in accordance with the plant modification process, which incorporates Detailed Control Room Design Review concepts.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 11

What are the changes that power uprate will have on the Safety Parameter Display System (SPDS).

Response to Question 11

The only changes to SPDS as a result of power uprate are those which result from setpoint changes. These are reflected in values at which system status flags and parameter values change color. The information presented on the SPDS display (top level display) and the method of presentation remain the same as before uprate. There are minor changes to lower level displays which support SPDS. Examples of these are the flags for SRV lift pressure change from 1110 psig to 1145 psig, and the flag for high RPV pressure scram changes from 1045 psig to 1080 psig. Limit displays for Boron Injection Initiation Temperature, Heat Capacity Temperature Limit, Pressure Suppression Pressure and SRV Tail Pipe Level Limit also have minor changes due to use of Reload 12/Cycle 13 parameters in the supporting calculation.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 12

Describe any changes the power uprate will have on the operator training program and the plant simulator.

Response to Question 12

- Training content has been revised to include the power uprate. FitzPatrick operator training is performed in accordance with NRC regulatory requirements, INPO and applicable industry guidance. Simulator training on the power uprate has been conducted, as identified in Question 9, and any subsequent issues identified as a result of the startup testing program will be incorporated in Licensed Operator training during 1997.

- Additionally, classroom training was conducted in 2 to 2 1/2 hour sessions, in an interactive mode, over a six week period during cycle 96-6. The training involved an overview of various aspects of the power uprate (e.g., parameter values changes, setpoint changes, startup test plan, etc.).

- The plant simulator is affected by changes in plant system parameters to reflect the uprate power operating condition, including changes made to control room instrumentation and the plant EPIC system (process computer system). These changes have been implemented on the Simulator and any discrepancies identified as a result of the startup testing program will be incorporated in accordance with simulator Configuration Management procedural requirements.

As requested, a license condition which requires a review of the results of the startup test program, to determine any potential effects on operator training, will be added to the facility Operating License (OL) as described in Attachment II.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 13

As indicated in GE report NEDC-32016P-1, the power uprate will cause a higher average neutron flux, which may increase the integrated fluence over the period of the plant life. As a result of the increase in the integrated fluence, the licensee must address the following issues:

1) Section 3.3.1 of the report indicates that the upper shelf energy (USE) of the vessel will be greater than 50 ft-lb at expiration of its license (EOL). However, during our review of the licensee's response to GL 92-01, the licensee could not demonstrate that the lower shell axial welds and circumferential beltline weld would remain above the Appendix G, 10 CFR Part 50, USE screening criteria of 50 ft-lb at EOL. The licensee submitted an equivalent margins analysis to demonstrate compliance with Appendix G, 10 CFR Part 50. The analysis was a generic analysis for BWRs that was prepared by GE.

   Is the previously submitted equivalent margins analysis topical report applicable to the FitzPatrick reactor vessel with the increased neutron fluence resulting from the power uprate? The licensee should compare the neutron fluence at power uprate EOL to the value in the topical report to demonstrate the analysis is bounding for the power uprate.

   OR

   The licensee should submit Charpy impact test data and analysis that demonstrates these welds will be above 50 ft-lb at EOL.

2) How will the power uprate affect their pressure temperature limits? Do the current P-T limits meet the requirements of Appendix G, 10 CFR Part 50? When will the P-T limits be updated to account for the revised neutron flux?

3) How will the power uprate affect their surveillance capsule withdrawal schedule and its compliance with Appendix H, 10 CFR Part 50?

Response to Question 13

Reactor Material Surveillance Schedule

In NRC Letter, Applicability of G.E. Topical Report NEDO-32205-A, Revision 1, February 1994, for the James A. FitzPatrick Nuclear Power Plant (TAC No. M89580), dated March 30, 1995, the NRC staff determined the analyses in the G.E. report are applicable to the FitzPatrick reactor vessel, and that the FitzPatrick reactor vessel will maintain margins of safety against fracture equivalent to those required by 10 CFR 50 Appendix G and the ASME code. NEDO-32205-A, Revision 1, is still applicable.

Response to Question 13 cont'd

10 CFR 50 Appendix H requires a material surveillance program to monitor changes in the fracture toughness of reactor pressure vessel ferritic materials. 10 CFR 50 Appendix H requires that reactor beltline materials surveillance programs must comply with ASTM E185, as modified by Appendix H.

The FitzPatrick TS commit to a surveillance program in accordance with ASTM E185-82 and 10 CFR 50 Appendix H. The current FitzPatrick pressure-temperature curves were generated based on Reg Guide 1.99, Revision 2. The first surveillance capsule was removed in April 1985 after 6 Effective Full Power Years (EFPY). The schedule for subsequent capsule removal was approved by the NRC.

The Authority plans to remove the second capsule during the 1996 refueling outage. This schedule is designed to support operation following the 1998 refueling outage. Specimens will be tested based upon ASTM E185-82 and 10 CFR 50 Appendix H.

The Authority plans to remove the third capsule some time late in plant life. The schedule for removal will be based upon data from the second capsule. In accordance with 10 CFR 50 Appendix H, a proposed withdrawal schedule will be submitted with a technical justification as specified in 10 CFR 50.4. This proposed schedule must be approved by the NRC prior to implementation.

There is a fourth capsule that is a spare and currently has no specific schedule for withdrawal. This capsule contains reconstituted specimens and was installed one operating cycle after the first capsule was removed. It will be considered for removal, based on test results from previous capsules. As stated above, such a schedule will be submitted to the NRC in accordance with 10 CFR 50 Appendix H.

Pressure-Temperature Limit Curves

The current pressure temperature curves were developed from General Electric Report DRF 137-0010, "Implementation of Regulatory Guide 1.99, Revision 2 for the James A. FitzPatrick Nuclear Power Plant", dated June 1989. These curves are valid through 16 EFPY.

General Electric has evaluated the applicability of the current curves under power uprate conditions (Reference 6). After uprate, but prior to reaching 16 EFPY the incremental exposure due to uprate will have an insignificant effect on the existing P/T curves, and these curves are still valid. The basis for this conclusion was comparing the changes in adjusted reference temperature (ART) and upper shelf energy (USE) assuming that the plant is operating at uprated power for the 32 EFPY life of the RPV. The change in ART was only 2.5°F and USE remained above 50 ft-lbs at end of life, therefore the effect of uprate on ART and USE is not significant.

Response to Question 13 cont'd

The applicability of NEDO-32205-A, Revision 1, will be determined and the Pressure-Temperature curves will be revised based upon the second capsule data and Regulatory Guide 1.99, Revision 2. A report will be submitted to the NRC, as specified in 10 CFR 50.4, within one year of the date of capsule withdrawal as required by 10 CFR 50 Appendix H. The expected date for submittal of the revised TS will be provided with the report. The Authority estimates that the new curves will not be used until the middle of 1999.

Upper Shelf Energy

The current design basis for end of life (EOL) fluence for FitzPatrick is 32 effective full power years (EFPY) based upon 40 years of power operation at 80% capacity factor. The predicted fluence for 32 EFPY was used to determine that vessel material properties meet the upper shelf energy (USE) requirements of 10 CFR 50 Appendix G. At the end of cycle 12, FitzPatrick has operated for less than 13.5 EFPY. Based upon future operation, FitzPatrick will not reach 32 EFPY, therefore previous EOL evaluations and the applicability of NEDO-32205-A, Revision 1, are still valid. The current estimate for EOL conservatively assumes 1.05 EFPY for each future year of operation (due to Uprate), 100% capacity factor, and a 45 day refueling outage every two years.

To summarize, Topical Report NEDO-32205-A, Revision 1, February 1994, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," is still applicable to the FitzPatrick reactor vessel and the existing P-T curves are conservative and valid for FitzPatrick Power Uprate implementation. In addition, ongoing plant specific analyses bound the effects of Power Uprate and thereby demonstrate compliance with 10 CFR 50, Appendix G.

Question 14

Is there a discrepancy between Section 4.2 of Reference 2 and UFSAR Section 6.4.2 regarding Automatic Depressurization System (ADS) initiation?

Response to Question 14

Section 4.2.4 of NEDC-32016P-1 (Reference 2) states that the plant design requires a minimum flow capacity for the SRVs, and that ADS initiates on low water level. Section 6.4.2 (Automatic Depressurization System) of the UFSAR describes actuation logic parameters for the ADS (i.e., override inhibit, water level, pump discharge pressure, and time delay).

The statement in Section 4.2.4 of NEDC-32016P-1 describes one of the plant safety design features of the ADS, which is described in UFSAR Section 6.4.2 and in UFSAR Section 4.4.4 (Pressure Relief System Safety Design Bases). This safety design feature is that ADS reduces the reactor coolant system pressure for small breaks in the event of a malfunction of the High Pressure Coolant Injection (HPCI) System so that the Low Pressure Coolant Injection (LPCI) and the Core Spray systems can operate. HPCI and ADS both initiate on low RPV water level, although the setpoints used are different (i.e., the ADS level setpoint is lower than the HPCI setpoint since ADS is a backup to HPCI).

Based on the above, the Authority concludes that the statement in Section 4.2.4 of NEDC-32016P-1 is correct. The statement, taken in context, should be interpreted that it describes a safety design bases of ADS and does not provide the specific RPV level setpoint used for ADS actuation or other design basis information.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

Question 15

What systems have been reviewed that are not affected or are minimally effected by the uprated power level?

Response to Question 15

Systems not Impacted by the Uprated Power Level

In addition to the systems listed in Table J-1 of NEDC-31897P-1 (Reference 4), the following systems have been evaluated and are not effected by the uprated power level:

- Automated Records Management
- Compressed Air
- Contaminated Equipment Vents
- Contaminated Laundry (not currently in use)
- Diesel Generator Room Ventilation
- Emergency Diesel Generators
- Fire Protection
- Gate House
- Hydrogen Storage and CO_2 Purge Supply
- Lube Oil
- Machine Shop Equipment
- Plumbing, Sanitary, and Lab Equipment
- Raw Water Treating
- Demineralized Water
- Sample System
- Screenwell and Pumphouse Ventilation
- Sewage Treatment
- Steam Seal System
- Tools and Servicing Equipment
- Warehouse Heating and Ventilation
- Yard City Water
- Radwaste Building Ventilation
- Stack and Stack Equipment

Systems Minimally Impacted by the Uprated Power Level

The following systems are minimally effected by the uprated power level:

- Hydrogen Water Chemistry
- Local Panels and Racks
- Process Computer
- Stator Water Cooling
- Extraction Steam

Question 16

Does the power uprate effect equipment qualifications of electrical equipment inside and outside containment?

Response to Question 16

The Authority evaluated the effect of operation at 2536 MW_t on the Environmental Qualification (EQ) Program for electrical equipment (Reference 7).

The effects of temperature, radiation and humidity during normal operation and of temperature, pressure, radiation and humidity for accidents were considered for environmentally qualified electrical equipment located within the drywell, reactor building, turbine building and steam tunnel. Accidents considered were loss of coolant, control rod drop, fuel handling, high energy line break and main steam line break as applicable to each area evaluated. New qualified life estimates were required for some components as a result of increased normal operating temperature.

From this evaluation, the Authority has concluded that environmental qualification is maintained for all environmentally qualified electrical components following power uprate. The Authority is in the process of updating EQ files to reflect the results of this evaluation. The EQ files will be updated prior to startup from R12/C13 RFO.

REQUEST FOR ADDITIONAL INFORMATION REGARDING POWER UPRATE

REFERENCES

1. Stone & Webster Engineering Corporation, "Core Power Uprate Engineering Report for James A. FitzPatrick Nuclear Power Plant," December 1991 (Engineering Report)

2. General Electric Report NEDC-32016P-1, "Power Uprate Safety Analysis for James A. FitzPatrick Nuclear Power Plant," April 1993 (proprietary), Including Errata and Addenda Sheet No. 1, dated January 1994

3. EPRI NP-1558, "A Review of Equipment Aging Theory and Technology," Final Report, September 1980

4. General Electric Report No. NEDC-31897P-1, "Licensing Topical Report - Generic Guidelines for GE BWR Power Uprate," dated June 1991 (proprietary)

5. JAF-RPT-MULTI-02420, "Power Uprate Startup Test Program Plan," dated August 28, 1996

6. General Electric Report No. GE-NE-B1301805-05R1, "Reactor Vessel Fracture Toughness Engineering Evaluation for the James A. FitzPatrick 104% Power Uprate," dated March 1996 (proprietary)

7. NYPA Report No. JAF-RPT-MISC-02512, "Evaluation of Power Uprate on the Environmental Qualification Program," dated October 4, 1996

Attachment II to JPN-96-043

REQUESTED CHANGES TO THE FACILITY OPERATING LICENSE

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

Add the following to the Operating License as Section 2.E:

E. Power Uprate License Amendment Implementation

The licensee shall complete the following actions as a condition of the approval of the power uprate license amendment.

(1) Recirculation Pump Motor Vibration

Perform monitoring of recirculation pump motor vibration during initial Cycle 13 power ascension for uprated power conditions.

(2) Startup Test Program

The licensee will follow a startup testing program, during Cycle 13 power ascension, as described in GE Licensing Topical Report NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate." The startup test program includes system testing of such process control systems as the feedwater flow and main steam pressure control systems. The licensee will collect steady-state operational data during various portions of the power ascension to the higher licensed power level so that predicted equipment performance characteristics can be verified. The licensee will do the startup testing program in accordance with its procedures. The licensee's approach is in conformance with the test guidelines of GE Licensing Topical Report NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," June 1991 (proprietary), GE Licensing Topical Report NEDO-31897, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," February 1992 (nonproprietary), and NEDC-31897P-A, Class III (proprietary), May 1992.

(3) Human Factors

The licensee will review the results of the Cycle 13 startup test program to determine any potential effects on operator training. Training issues identified will be incorporated in Licensed Operator training during 1997. Simulator discrepancies identified will be addressed in accordance with simulator Configuration Management procedural requirements.

Attachment III to JPN-96-043

TS PAGE 254c CHANGES

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification 3.5.H;
- The Minimum Critical Power Ratio (MCPR) and MCPR low flow adjustment factor, K_f, of Specifications 3.1.B and 4.1.E;
- The Linear Heat Generation Rate (LHGR) of Specification 3.5.I;
- The Reactor Protection System (RPS) APRM flow biased trip settings of Table 3.1-1;
- The flow biased APRM and Rod Block Monitor (RBM) rod block settings of Table 3.2-3; and
- The Power/Flow Exclusion Region of Specification 3.5.J.

and shall be documented in the Core Operating Limits Report (COLR).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13, August 1996.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, Revision 2, April 1993.
3. "BWP Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1991.
4. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992.

Attachment IV to JPN-96-043

MARK-UP OF TS PAGE 254c

NOTE 1: Deletions are shown in ~~strikeout~~, and additions are in **bold.**

NOTE 2: Previous amendment revision bars are shown and will be deleted.

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification 3.5.H;
- The Minimum Critical Power Ratio (MCPR) and MCPR low flow adjustment factor, K_f, of Specifications 3.1.B and 4.1.E;
- The Linear Heat Generation Rate (LHGR) of Specification 3.5.I;
- The Reactor Protection System (RPS) APRM flow biased trip settings of Table 3.1-1;
- The flow biased APRM and Rod Block Monitor (RBM) rod block settings of Table 3.2-3; and
- The Power/Flow Exclusion Region of Specification 3.5.J.

and shall be documented in the Core Operating Limits Report (COLR).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P**-A-13**, ~~latest approved version and amendments.~~ **August, 1996.**

2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, ~~October, 1986 including latest errata and addenda.~~ **Revision 2, April 1993**

~~3. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.~~

~~4~~**3**."BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, June 1991.

~~5~~**4**. "BWR Owners' Group Long-term Stability Solutions Licensing Methodology," NEDO-31960-A, Supplement 1, March 1992.

Summary of Commitments

Number	Commitment	Due Date
JPN-96-043-01	Update the FAC models and perform computer analyses to include the inspection data from R12/C13 and the power uprate heat balance inputs. These updates are required as part of plant procedures. Inspection points for the next RFO (i.e., R13/C14 - which is currently scheduled to commence in the Fall 1998) will be based on these new analyses.	Prior to the R13/C14 RFO.
JPN-96-043-02	Ensure terminal voltage operating guidelines are in place to reflect the power uprate.	Following approval of the Authority's power uprate submittal.
JPN-96-043-03	The applicability of NEDO-32205, Revision 1, will be determined and the Pressure-Temperature curves will be revised based upon the second capsule data and Regulatory Guide 1.99, Revision 2. A report will be submitted to the NRC as specified in 10 CFR 50.4. The expected date for submittal of the revised TS will be provided with the report.	Within one year of the date of capsule withdrawal, as required by 10 CFR 50, Appendix H.
JPN-96-043-04	Update the EQ files to reflect the power uprate.	Prior to startup from R12/C13 RFO.