SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION OF THE

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

U. S. Atomic Energy Commission
Directorate of Licensing
Washington, D. C.

Issue Date: February 1, 1973

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Appendices

I. INTRODUCTION

The Safety Evaluation Report (SER) for the James A. FitzPatrick Nuclear Power Plant was issued on November 20, 1972. The SER identified matters requiring additional information from the applicant. The Advisory Committee on Reactor Safeguards' letter to the Chairman of the USAEC of December 15, 1972 also requested additional information on some specific concerns.

The purpose of this supplement is to update the SER based on the Staff review of information regarding the issues requiring additional evaluation submitted since November 20, 1972 and to discuss our evaluation and implementation of the recommendations from the ACRS letter.

The sections of the SER which have been affected by these reviews have been amended herein and are numbered to coincide with the applicable sections of the SER.

In addition, the supplement contains an updated chronology as Appendix A, the report of the ACRS as Appendix B and an errata to the SER as Appendix C.

II. REVISED SECTIONS OF SAFETY EVALUATION REPORT

3.2 Nuclear Design

We have started a review of fuel densification and its effect on reactor operation, including transients and postulated loss-of-coolant accidents. The Staff's preliminary investigations and conclusions are reported in "Technical Report on Densification of Light Water Reactor Fuels" dated

November 14, 1972. This report concludes that irradiation-induced densification of fuel in all light water reactors may occur and that it should be assumed that densification and the resulting formation of fuel column gaps occur immediately when the reactor is operating at substantial power. The report also provides the essential elements to be included in calculational models used to account for the effects of fuel densification.

We have requested the applicant to provide the necessary analyses and other relevent data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident, using the guidance provided in the Staff's report.

The applicant's response is contained in a generic report prepared by the General Electric Company, NEDM-10735, "Densification Considerations in BWR Fuel Design and Performance" dated December 1972. We are presently reviewing this report. If our review of the analysis presented indicates that changes in design or operating conditions are necessary to maintain required margins, we will require appropriate limits of operating power and time on the initial fuel loading.

3.4 Reactivity Control

This section of the SER indicated that the applicant was

considering using a Rod Sequence Control System (RSCS) in addition to the rod worth minimizer (RWM) to reduce the probability of high worth rods being involved in a Control Rod Drop Accident.

The RSCS is a hard wired system which is electrically independent from the RWM and utilizes inputs from the full-in and full-out switches in the rod position indicator probes and rod sequence selector switches. It wires the rod select relays into groups that control four sequence patterns. A relay either inhibits or permits movement of all of the rods assigned to a sequence pattern.

The RSCS independently restricts the selection and withdrawal of control rods up through 50% rod density (checkerboard pattern with 50% of the control rods full-out and 50% full-in).

At power levels exceeding 50% rod density, the applicant states that the consequences of a Control Rod Drop accident using the maximum rod worth calculated at a fuel exposure of 6500 MWD/t indicate that the peak fuel enthalpy is below the threshold value (280 cal/gm) assumed to cause rapid fuel dispersal and damaging pressure pulses to the reactor core and that the radiological doses at the site boundary from the estimated fuel cladding failures are well within the guidelines of 10 CFR Part 100. The applicant has indicated that the proposed RSCS will be installed prior to operation above 1% of rated power.

Our evaluation of the RSCS with respect to the Control Rod Drop Accident is discussed in Section 10.4 of this supplement.

4.2 Reactor Coolant Pressure Boundary Design Evaluation

In connection with tests that will be made of piping responses to transients during preoperational testing, the applicant has stated that if excessive vibration is visually observed, measurements of deflections will be made and the system deflections and stresses will be limited to the acceptance limits of FSAR Appendix C by the addition of hangers or restraints as required.

4.4 Reactor Recirculation System

In connection with Staff and ACRS concerns regarding recirculation pump and motor overspeed, the applicant has agreed, in his letter of January 11, 1973, to install a decoupling clutch between the motor and pump during the first refueling outage and to perform a review of the piping arrangement inside the primary containment for possible missile protection of the containment. This review will be completed within a few months.

The Staff believes that the small probability of the postulated event occurring during the interim period makes full power operation acceptable prior to installation of the required protection at the first refueling outage.

4.11 Inservice Inspection Program

In response to an ACRS recommendation, PASNY has reconfirmed its intention to keep abreast of developments in automated equipment for inspection at presently inaccessible locations. This commitment is made in the PASNY letter of January 11, 1973.

5.2.3 Containment Isolation

On the basis of Staff calculations of the effects of main steam line isolation valve leakage and a continuing concern by the ACRS, PASNY was requested to submit plans for installation of a supplementary sealing system. PASNY has committed, in a letter dated January 11, 1973, to demonstrate and install an acceptable sealing system at the time of the plant's first refueling outage.

The Staff has completed an analysis of the doses which would result, should the design basis loss-of-coolant accident occur in the interim before the sealing system is installed. Using conservative assumptions, it has been found that with holdup of the leakage in intact steam lines and mixing in the condenser, the resulting radiological doses at the LPZ and the exclusion distance are below the limits of 10 CFR 100. Without mixing and dilution in the condenser, the limits of these guidelines could be exceeded. Administrative measures would be necessary to mitigate the thyroid dose to control room occupants to within the guidelines of General Design Criterion 19.

The Staff believes that the small probability of the postulated events, concurrent with seismic damage to the main steam line and condenser system during the interim period, coupled with the conservative nature of the analyses, and the feasibility of administrative measures to protect control room occupants, make it acceptable to operate the plant at full power power prior to the first refueling outage.

6.2.5 Discussion of ECCS Review

PASNY has submitted in Amendment 18 the calculated effects of fuel loading errors on peak fuel cladding temperatures under LOCA conditions. The temperature increases for the following errors are:

errors are:	ΔT, °F
(a) improper bundle location	40
(b) improper bundle orientation	60
(c) error in gadolinia concentration	135
(d) error in gadolinia rod position	160

Based on our review of the proposed quality assurance plan for the control of fuel enrichments and burnable poision, as discussed in Section 3.2 of the SER, we have determined that errors (a) and (b) above are improbable and errors (c) and (d) are highly improbable. Nevertheless, given that any of these errors occur, the resulting peak fuel cladding temperatures would still be below the 2300°F criterion.

6.4 Other Engineered Safety Features

We have reviewed the design and analysis of the Containment Atmosphere Dilution (CAD) system and find it acceptable. The evaluation was performed on the basis of the assumptions given in Safety Guide 7. In this regard, it has been demonstrated that with an initial oxygen level in the containment of 4%, initiation of the CAD system would not have to occur for at least 12 hours after an assumed design basis loss-of-coolant accident. To ensure that the containment repressurization pressure is limited to a value substantially below design pressure, the applicant has proposed a repressurization limit of 37 psig. The applicant has calculated that with no containment leakage, purging beginning at 31 days at a rate of 5 scfm would maintain containment pressure at 37 psig. We have calculated the site boundary dose from the 5 scfm purge rate between 31 and 61 days as 10 rem thyroid and less than 500 mrem whole body. We conclude that the containment repressurization limit is acceptable and that there is reasonable assurance that this limit can be maintained for acceptable operation of the CAD system in the unlikely event of a LOCA.

7.5 Anticipated Transients Without Scram (ATWS)

In response to an ACRS recommendation, the applicant has reconfirmed that he will install a recirculation pump trip prior to the start of commercial power operation. This is contained in the PASNY letter of January 11, 1973.

The Staff agrees that the addition of the recirculation pump trip as proposed by the applicant represents a substantial improvement in protection of the reactor for anticipated transients without scram; however, the Staff has not completed its review of all the transients as discussed in the General Electric Company Topical Report NEDO-10349. Completion of our review of this topic is pending receipt of and review of responses to additional information which was requested from General Electric in a letter dated June 13, 1972.

The pump trip is automatic on either a signal of high pressure or low water level. The pressure and level devices used for the pump trip are not the same level and pressure devices that are used for scram. In addition, the pressure switches used to trip the recirculation pumps will be of a different type and manufacturer than those used for scram. The Staff has not concluded that the proposed recirculation pump trip provides a completely acceptable degree of protection against anticipated transients without scram for reactors of

this general type. This conclusion is pending our receipt and review of the outstanding information cited above. The General Electric Company has indicated that the information requested by the Staff regarding anticipated transients without scram will be submitted as a topical report in early 1973.

7.6.2 Environmental Qualification

All environmental testing has been successfully completed with the exception of the solenoid valves for the safety/relief valves. Testing of the solenoid valves is still in progress.

7.7.1 Offsite Power

PASNY has submitted information in Amendment 18 concerning inadvertent isolation of the FitzPatrick plant from offsite power. This was indicated as an area of staff concern in the SER.

Modifications have been made to the circuitry which preclude any single failure from completely isolating the FitzPatrick plant. We have evaluated these modifications and have determined that they conform to General Design Criterion 17. The offsite power system is thus acceptable.

8.2.2 Gaseous Radwaste System

The Staff has calculated an infant thyroid dose of 11 mrem/year due to milk consumption from the nearest dairy

farm located about 1.2 miles south of the plant. Although this is somewhat greater than the 5 mrem/year guideline of proposed Appendix I to 10 CFR Part 50, we will not require the applicant to make any facility modifications at the present time. The Technical Specifications will require a comprehensive milk monitoring program to demonstrate that the actual dose does not exceed 5 mrem/year. In the event the measurements indicate that this dose could be exceeded, immediate and positive steps will be required to reduce it.

9.2.4 Seismic Design Criteria

The SER indicated that the applicant had applied an empirical factor of 1.3 to the peak resonant response in his response spectrum approach for design of equipment. At the Staff's request, the most highly stressed components have been reanalyzed using a factor of 1.5. The applicant has stated that these analyses are completed and the results have been evaluated for all but one component. The stresses remain within the allowable limits described in Section 9.3 of the SER with an adequate margin of safety.

9.3 Evaluation of Class I Structures and Equipment

PASNY has submitted a summary of the test methods and results used in the design of seismic Class I mechanical equipment such as fans, pump drives and valve operators. Analyses were made for equipment which was not tested.

The applicant has stated that an evaluation of the analyses made for the Class I structure and equipment verifies that all seismic Class I equipment is capable of withstanding all design seismic vibratory loading conditions. Further, the evaluation verifies that such equipment will remain functional under these conditions, as required.

10.4 Control Rod Drop Accident

The SER indicated the applicant's intent to install a Rod Sequence Control System (RSCS) as a backup to the Rod Worth Minimizer. The ACRS report also makes reference to this subject. In Amendment 18, PASNY provides additional information. The RSCS has been described in Section 3.4 of this supplement. It is designed to prevent the operator from withdrawing an out-of-sequence control rod during startup or shutdown while in the 100% to 50% rod density range, where rod density is defined as the percent of control rods fully inserted in the core. Beyond 50% rod density, General Electric has provided an estimate of the peak fuel enthalpy, which is 276 calories per gram for the postulated rod drop accident using the maximum rod worth calculated at a fuel exposure of 6500 MWD/T.

General Electric Company, in meetings with the Regulatory Staff, indicated that the consequences of the rod drop accident include cladding perforations in approximately 600 fuel rods (fuel rods that have peak fuel enthalpies greater than 170 calories per gram). Assuming 600 fuel rod perforations and the assumptionsgiven in Section 10.4 of the Staff's Safety Evaluation dated November 20, 1972, the dose consequences reported on page 10-6 of the Safety Evaluation would increase by a factor of two to a value of 2 rem thyroid at the nearest site boundary. This dose is still less than the dose for the design basis loss-of-coolant accident and significantly less than the guideline values of 10 CFR Part 100.

We have requested final electrical schematic drawings and additional information on quality assurance and control for our review. The review of the proposed system cannot be completed by the Staff until these details are submitted and the applicant confirms that the peak fuel enthalpies resulting from the postulated rod drop accident beyond 50% rod density do not exceed 280 calories per gram. This confirmation is anticipated by March 1973. In the event the finalized calculations for the peak fuel enthalpies at the 6500 MWD/T exposure exceed our acceptance criterion of

280 calories per gram, we will require the applicant to provide additional modifications to satisfy this criterion.

The Technical Specifications will require that the RSCS, when accepted and installed, be operable below 10% power level and that the control rod scram time (to 90% insertion) be less than 4.0 seconds, the value used in the revised rod drop accident analysis.

10.5 Main Steam and Feedwater Line Break Accident

The Staff's continuing review of reactor power plant safety indicates that the consequences of postulated pipe failures outside of the containment structure, including the rupture of a main steam or feedwater line, need to be adequately documented and analyzed by licensees and applicants, and evaluated by the Staff.

In the FitzPatrick plant, the main steam lines and feedwater lines are enclosed in a reinforced concrete tunnel from the drywell through the reactor building, opening directly to the turbine building. The control room and emergency diesel generator rooms are remote from these pipes and the vital switchgear room above the tunnel is protected by a 5 ft. thick reinforced concrete floor. Conventional quality assurance standards were used for steam and feedwater piping outside the primary containment.

Although it appears that the plant design is capable of withstanding the effects of a postulated rupture in the steam or

feedwater lines, we have requested the applicant to provide analyses and other relevant information needed to determine the consequences of such an event.

If the results of these analyses and the subsequent Staff review indicate that changes in the design of structures, systems, or components are necessary to assure safe reactor shutdown in the event this postulated accident situation should occur, design modifications of the FitzPatrick plant to accommodate the postulated failures described above will be required.

APPENDIX A
UPDATED CHRONOLOGY

November 14, 1972	Commission's Draft Environmental Statement issued.
November 20, 1972	Commission's Safety Evaluation issued.
November 20, 1972	AEC letter requests information on effects of fuel densification.
November 24, 1972	PASNY submits letter indicating that a Rod Sequence Control System will be provided to augment the Rod Worth Minimizer in partial response to the AEC letter of 10/31/72.
November 25, 1972	ACRS Subcommittee meeting and site visit.
November 30, 1972	AEC letter requests additional information and clarification on proposed containment atmosphere control system.
December 7, 1972	ACRS full committee meeting.
December 8, 1972	AEC letter provides guidance to PASNY with respect to nondestructive testing requirements of fittings in the reactor coolant pressure boundary.
December 15, 1972	ACRS letter received.
December 18, 1972	AEC letter requests analysis of pipe failures outside the containment structure.
December 21, 1972	AEC letter transmits ACRS letter to PASNY and requests action on several items.
December 22, 1972	PASNY submits Amendment 18 together with Supplements Nos. 17 and 18 to the FSAR containing a cross reference index and supplementary information.
December 26, 1972	PASNY submits letter containing additional CAD system information in response to the AEC letter of 11/30/72.

December 27, 1972	PASNY submits copies "as-built" control, instrumentation and electrical system drawings which were reviewed by the staff.
January 5, 1973	PASNY submits response to AEC letter of 11/20/72 on fuel densification.
January 8, 1973	PASNY submits Amendment 20 together with Supplement No. 19 to the FSAR containing supplementary and updated information. PASNY also submits letter committing to a date for response to information requested in AEC letter of 12/18/72.
January 10, 1973	PASNY submits Amendment 19 to the application containing updated information of a non-technical nature.
January 15, 1973	PASNY submits response to AEC letter of 12/21/72.
January 22, 1973	PASNY submits additional information on CAD system.
January 26, 1973	Meeting with PASNY to obtain additional information for qualitative assessment of high energy fluid system failure outside containment.

Appendix B

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 15, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON JAMES A. FITZPATRICK NUCLEAR POWER PLANT

Dear Dr. Schlesinger:

At its 152nd meeting, on December 7-9, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application by the Power Authority of the State of New York (PASNY) and the Niagara Mohawk Power Corporation for authorization to operate the James A. FitzPatrick Nuclear Power Plant at power levels up to 2436 MW(t). This project was considered at a Subcommittee meeting at the plant site on November 25, 1972. During its review, the Committee had the benefit of discussions with representatives and consultants of the Power Authority of the State of New York, the Niagara Mohawk Power Corporation, the General Electric Company, the Stone and Webster Engineering Corporation, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee reported to the Commission on the construction of this plant in its letter of January 27, 1970.

The James A. FitzPatrick Nuclear Power Plant is located on the shore of Lake Ontario approximately seven miles northeast of Oswego, New York, the nearest population center, and 36 miles northwest of Syracuse. The plant employs a General Electric boiling water nuclear steam supply system of design similar to several previously approved for operation.

PASNY owns the nuclear plant and has sole responsibility for its construction. The Niagara Mohawk Power Corporation will operate the FitzPatrick Plant under contract with PASNY. Niagara Mohawk also owns and operates the Nine Mile Point Nuclear Station which is located 3000 ft. west of the FitzPatrick Plant on a contiguous site.

The FitzPatrick reactor will operate at a power density similar to that of the recently reviewed Browns Ferry and Peach Bottom reactors. The reactor core design has been modified from that proposed at the construction permit stage. The originally planned boron-steel control curtains have been replaced by full-length gadolinia-urania fuel pins and by part-length gadolinia-urania fuel pins to provide axial flux shaping throughout the fuel cycle.

The FitzPatrick Plant reactor is the first to be placed in service with the proposed internals arrangement. An acceptable prototype vibration testing program has been committed to by the applicants.

Analyses of postulated control rod drop accidents have been revised by the applicants to employ a more realistic rate of reactivity insertion than formerly assumed, and to account for changes made in core design, in the use of a number of fuel enrichments, and employment of full-and part-length gadolinia bearing fuel rods. These analyses indicate that, for postulated accidents during certain portions of the fuel cycle, the results are unacceptable. The applicants have proposed a rod sequence control system which is intended to render the probability of occurrence of such an accident negligibly low. This matter is under review and should be resolved in a manner satisfactory to the Regulatory Staff and the Committee. Approved measures should be placed into effect prior to operation above 1% of rated power.

The potential effects of some aspects of fuel performance and LOCA-related phenomena on acceptable linear fuel heat ratings for the FitzPatrick Plant are under study. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

For control of combustible gas concentrations in the containment following a postulated loss-of-coolant accident, the applicants will install a containment atmospheric dilution (CAD) system by the end of the first normal refueling period. With this system the desired dilution is accomplished by controlled addition of nitrogen, and results in the maintenance of higher containment pressure than would otherwise exist during a portion of the post-LOCA period. The Committee believes that, in general, use of such dilution schemes, which involve repressurization of the containment, is not desirable. However, as an added provision to a plant well along in construction, use of this approach is believed by the Committee to be acceptable. The Committee nevertheless recommends that the applicants study means to assure that the peak repressurization pressure will be limited to a value substantially below the containment design pressure.

The in-service inspection program proposed for the reactor coolant system complies with Section XI of the ASME Boiler and Pressure Vessel Code to the extent permitted by the existing design. The Committee believes the program is acceptable, but recommends that the applicants continue to study means of assuring reactor vessel integrity in regions currently inaccessible for inspection.

The applicants will employ a recirculation pump trip as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The Committee believes that this represents a substantial improvement and should be provided prior to start of commercial power operation. The specific means for implementing the pump trip should be resolved in a manner satisfactory to the Regulatory Staff.

In the unlikely event that a break occurs in the recirculation pump discharge line, the pump impeller may act as a turbine causing the pump and motor to overspeed and become potential sources of missiles. The applicant is reviewing means of dealing with this matter, including an overriding clutch between the pump and motor. The Committee believes that this matter should be resolved in a manner satisfactory to the Regulatory Staff.

In its construction report on the FitzPatrick Plant the Committee noted that additional information should be provided by the applicant regarding leakage through the main steam line isolation valves. Recent boiling water reactor plants reviewed for construction have included provisions for sealing the isolation valves. The Committee believes 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. This matter should be resolved in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the James A. FitzPatrick Nuclear Power Plant can be operated at power levels up to 2436 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References attached.

References

1. Final Safety Analysis Report, Volumes 1-5 to the James A. FitzPatrick Nuclear Power Plant

2. Supplements 1-16 to the Final Safety Analysis Report

3. Power Authority of the State of New York letter dated October 2, 1972

4. Power Authority of the State of New York letter dated November 1, 1972

5. Power Authority of the State of New York letter dated November 2, 1972

6. Safety Evaluation, dated November 20, 1972, by the Directorate of Licensing

APPENDIX C

ERRATA TO SAFETY EVALUATION REPORT

1. Page 1-1 line 13 - change "2550 Mwt" to 2535 Mwt"
2. Page 1-1 line 21 - add "with safety significance" after "components"
3. Page 1-5 line 16 - delete one "i" in information.
4. Page 2-6 line 3 - add "approximately" before "5 miles."
5. Page 2-7 line 14 - change "249" to "248"
6. Page 2-10 line 20 - change "232.8" to "222.8"
7. Page 2-10 line 21 - add "which extends" after "tunnel"
8. Page 2-11 line 11 - change "4×10^2" to "40"
9. Page 2-11 line 11 - add "at the Oswego Public Water Supply Intake," after "10^3,"
10. Page 2-11 line 11 - change "(lesser values)" to (least dilution)"
11. Page 2-13 line 2 - add "which was" after "fault," and add "earlier" after "disclosed."
12. Page 2-14 line 5 - change "lake." to "lake or eastward along the lake shore."
13. Page 3-6 lines 5, 6 - change "based on....(Docket Nos. 50-359/360/396)," to "the applicant has shown that"
14. Page 3-6 lines 10/11 - change "based on....plant." to "considering its low probability and the fact that the consequences are not unduly severe."
15. Page 3-13 line 1 - add "Appendix C of the FSAR" after "with"
16. Page 3-13 line 1 - Insert "1965" before "ASME"
17. Page 3-13 line 1 - add phrase "was used as a guide." after "code."

18. Page 3-13 line 3 - insert phrase "in accordance with the seismic design conditions described in Appendix C of the FSAR." after "are designed"

19. Page 3-14 line 13 - change "member to the...static load." to "member by means of a system dynamic analysis."

20. Page 4-1 line 9 - change "reactor building" to read "primary containment"

21. Page 4-2 line 13 - add "and" after "designed,"

22. Page 6-6 line 11 - change "reactor core....LPCIS." to read "reactor core in the event of a large coolant line break (from 0.1 to 4.2 ft^2)."

23. Page 6-7 line 3 - change "both" to read "the"

24. Page 7-5 line 18 - insert the word "coincident" after "tripped on" Insert the phrase "and low water level." after word "pressure"; delete "only."

25. Page 7-10 line 14 - change "two" to "four"

26. Page 7-10 line 15 - change ", one" to ", two"; change "each tank" to read "each pair of tanks"

27. Page 8-15 last line - eliminate "no"

28. Page 11-1 line 1 - insert "combined" after "The" and "Nine Mile Point Nuclear Station Unit 1" after (JAFNNP)

29. Page 11-1 line 3 - insert "Station" after "Nuclear"

30. Page 11-2 line 6 eliminate "excluding instrumentation,"

31. Page 11-2 line 19 - insert "Point" after "Mile"

32. Page 11-2 lines 22/23 - change "Niagara Mohawk....New York." to read "Power Authority of the State of New York and consultants as required."

33. Page 11-4 line 19 - insert following sentence after "modifications." "The PASNY Resident Engineer is a member of the Site Operations Review Committee."

34. Page 12-1 line 9 - eliminate "and testing"

35. Page 12-1 lines 15/16 - eliminate "as a member....committee,"