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August 25, 1980

Docket No. 50-364

Director
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. A. Schwencer

RE: JOSEPH M. FARLEY NUCLEAR PLANT-UNIT NO. 2
COMPLIANCE WITH CODE OF FEDERAL REGULATIONS

Gentlemen:

Enclosed is an itemized review of the Farley Nuclear Plant - Unit 2 compliance with applicable 10CFR, Parts 20, 50 and 100 regulations. It is Alabama Power Company's position that Farley Nuclear Plant - Unit 2 complies with the applicable regulations except in those cases where specific exemptions have been justified and approved by the staff. We base our confidence in this conclusion on the references in the enclosure, plus the lengthy review and licensing process that the Farley Nuclear Plant Unit 2 has undergone. The design process of our own employees, the quality assurance programs of Alabama Power Company and the NSSS vendor, the independent review of the NRC staff and ACRS, and the additional independent review of the Atomic Safety and Licensing Boards provide reasonable assurance that the public health and safety will be protected.

If you have any questions, please advise.

Very truly yours,

F. L. Clayton, Jr.

F. L. Clayton, Jr.

FLCJr/TNE:aw

Enclosure

cc: Mr. R. A. Thomas
Mr. G. F. Trowbridge

SWORN TO AND SUBSCRIBED BEFORE
ME THIS ____ DAY OF ______________,
1980.

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Notary Public

My Commission Expires: ____________

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ENCLOSURE

COMPLIANCE OF FARLEY NUCLEAR PLANT UNIT 2 WITH THE NRC REGULATIONS
OF 10 CFR PARTS 20, 50, AND 100

Regulation (10 CFR)	Compliance
20.1(a)	This regulation states the general purposes for which the Part 20 regulations are established and does not impose any independent obligations on licensees.
20.1(b)	This regulation describes the overall purpose of the Part 20 regulations to control the possession, use and transfer of licensed material by any licensee, such that the total dose to an individual will not exceed the standards prescribed therein. It does not impose any independent obligations on licensees.
20.1(c)	Conformance to the ALARA principle stated in this regulation is ensured by the implementation of APC policies and appropriate Technical Specifications and health physics procedures. Chapters 11 and 12 of the FSAR describe the specific equipment and design features utilized in this effort.
20.2	This regulation merely establishes the applicability of the Part 20 regulations and imposes no independent obligations on those licensees to which they apply.
20.3	The definitions contained in this regulation are adhered to in all appropriate Technical Specifications and procedures, and in applicable sections of the FSAR.
20.4	The Units of Radiation Dose specified in this regulation are acceptable and conformed to in all applicable FNP procedures.
20.5	The Units of Radioactivity specified in this regulation are accepted and conformed to in all applicable FNP procedures.
20.6	This regulation governs the interpretation of regulations by the NRC and does not impose independent obligations on licensees.
20.7	This regulation gives the address of the NRC and does not impose independent obligations on licensees.

20.101	The radiation dose limits specified in this regulation are complied with through the implementation of and adherence to administrative policies and controls and appropriate health physics procedures developed for this purpose. Conformance is documented by the use of appropriate personnel monitoring devices and the maintenance of all required records. (See FSAR Chapter 12)
20.102	When required by this regulation, the accumulated dose for any individual permitted to exceed the exposure limits specified in 20.101(a) is determined by the use of Form NRC-4. Appropriate health physics procedures and administrative policies control this process. (See FSAR Chapter 12)
20.103(a)	Compliance with this regulation is ensured through the implementation of appropriate health physics procedures relating to air sampling for radioactive materials, and bioassay of individuals for internal contamination. Administrative policies and controls provide adequate margins of safety for the protection of individuals against intake of radioactive materials. The systems and equipment described in Chapters 11 and 12 of the FSAR provide the capability to minimize these hazards.
20.103(b)	Appropriate process and engineering controls and equipment, as described in Chapters 11 and 12 of the FSAR, are installed and operated to maintain levels of airborne radioactivity as low as practicable. When necessary, as determined by FNP administrative guidelines, additional precautionary procedures will be utilized to limit the potential for intake of radioactive materials.
20.103(c)	The FNP Respiratory Protection Procedure implements the requirements of this regulation by ensuring the proper use of approved respiratory protection equipment. The FNP Respiratory Protection Procedure incorporates fully the stipulations of Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection." (See FSAR Chapter 12)
20.103(d)	This regulation describes further restrictions which the Commission may impose on licensees. It does not impose any independent obligations on licensees.
20.103(e)	The notification specified by this regulation was made as required on February 28, 1977.
20.103(f)	The Respiratory Protection Program is in full conformance with the requirements of 20.103(c).
20.104	Conformance with this regulation is assured by appropriate APC policies regarding employment of individuals under the age of 18 and the FNP Health Physics Manual restricting these individuals' access to restricted areas.

20.105(a)	Chapters 11 and 12 of the FSAR provide the information and related radiation dose assessments specified by this regulation.
20.105(b)	The radiation dose rate limits specified in this regulation will be complied with through the implementation of FNP procedures, Technical Specifications, and administrative policies which control the use and transfer of radioactive materials. Appropriate surveys and monitoring devices will document this compliance.
20.106(a)	Conformance with the limits specified in this regulation will be assured through the implementation of FNP procedures and applicable Technical Specifications which provide adequate sampling and analyses, and monitoring of radioactive materials in effluents before and during their release. The level of radioactivity in station effluents will be minimized to the extent practicable by the use of appropriate equipment designed for this purpose, as described in Chapter 11 of the FSAR.
20.106(b) 20.106(c)	APC has not and does not currently intend to include in any license or amendment applications proposed limits higher than those specified in 20.106(a), as provided in these regulations.
20.106(d)	Appropriate allowances for dilution and dispersion of radioactive effluents will be made in conformance with this regulation, and are described in detail in Chapter 11 of the FSAR, and in appropriate reports required by the Technical Specifications.
20.106(e)	This regulation provides criteria by which the Commission may impose further limitations on releases of radioactive materials made by a licensee. It imposes no independent obligations on licensees.
20.106(f)	This regulation merely states that the provisions of 20.106 do not apply to disposal of radioactive material into sanitary sewerage systems. It imposes no independent obligations on licensees.
20.107	This regulation merely clarifies that the Part 20 regulations are not intended to apply to the intentional exposure of patients to radiation for the purpose of medical diagnosis or therapy. It does not impose any independent obligations on licensees.
20.108	Necessary bioassay equipment and procedures, including Whole Body Counting, will be utilized to determine exposure of individuals to concentrations of radioactive materials. Appropriate health physics procedures and administrative policies implement this requirement. (See FSAR Chapter 12)

20.201 The surveys required by this regulation will be performed at adequate frequencies and contain such detail as to be consistent with the radiation hazard being evaluated. When necessary, the Radiation Special Work Permit system established at the station provides for detailed physical surveys of equipment, structures and work sites to determine appropriate levels of radiation protection. The FNP Health Physics Manual and applicable health physics procedures require these surveys and provide for their documentation in such manner as to ensure compliance with the regulations of 10 CFR Part 20. (See FSAR Chapter 12)

20.202(a) The FNP Health Physics Mar l and applicable health physics procedures set forth polic..s and practices which ensure that all individuals are supplied with, and required to use, appropriate personnel monitoring equipment. The Radiation Work Permit system will be established on Unit 2 to provide additional control of personnel working in radiation areas and to ensure that the level of protection afforded to these individuals is consistent with the radiological hazards in the work place. (See FSAR Chapter 12)

20.202(b) The terminology set forth in this regulation is accepted and conformed to in all applicable FNP procedures, Technical Specifications, and those portions of the FNP Health Physics Manual in which its use is made.

20.203(a) All materials used for labeling, posting, or otherwise designating radiation hazards or radioactive materials, and using the radiation symbol, conform to the conventional design prescribed in this regulation per FNP procedures.

20.203(b) This regulation will be conformed to through the implementation of appropriate health physics procedures and portions of the FNP Health Physics Manual relating to posting of radiation areas, as defined in 10 CFR Part 20.202(b)(2).

20.203(c) The requirements of this regulation for "High Radiation Areas" will be conformed to by the implementation of the FNP Technical Specifications and appropriate plant health physics procedures, as well as the FNP Health Physics Manual. The controls and other protective measures set forth in the regulation are maintained under the surveillance of the FNP Health Physics group.

It should be noted that Technical Specification 6.12.1 will provide alternate access control methods to be applied "in lieu of the 'control device' or 'alarm signal' required by paragraph 20.203(c)(2) of 10 CFR 20," which will prevent unauthorized entry into a high radiation area.

20.203(d) Each Airborne Radioactivity Area, as defined in this regulation, is required to be posted by provisions of the FNP Health Physics Manual and appropriate health physics procedures. These procedures also provide for the surveillance requirements necessary to determine airborne radioactivity levels.

20.203(e) The area and room posting requirements set forth in this regulation pertaining to radioactive materials are complied with through the implementation of appropriate health physics procedures, and portions of the FNP Health Physics Manual.

20.203(f) The container labeling requirements set forth in this regulation are complied with through the implementation of appropriate health physics procedures, and portions of the FNP Health Physics Manual.

20.204 The posting requirement exceptions described in this regulation are used where appropriate and necessary at FNP. Adequate controls are provided within the FNP health physics procedures to ensure safe and proper application of these exceptions.

20.205 All of the requirements of this regulation pertaining to procedures for picking up, receiving, and opening packages of radioactive materials are implemented by the FNP Health Physics Manual and appropriate health physics procedures. These procedures also provide for the necessary documentation to ensure an auditable record of compliance. (See FSAR Chapter 12)

20.206 The requirements of 10 CFR 19.12 referred to by this regulation are satisfied by the radiation worker training conducted at FNP. Appropriate health physics procedures set forth requirements for all radiation workers to receive this instruction on a periodic basis. (See FSAR Chapter 12)

20.207 The storage and control requirements for licensed materials in unrestricted areas are conformed to and documented through the implementation of FNP health physics procedures and applicable portions of the FNP Health Physics Manual. (See FSAR Chapter 12)

20.301 The general requirements for waste disposal set forth in this regulation are complied with through FNP health physics procedures, the Technical Specifications, and the provisions of the station license. Chapter 11 of the FSAR describes the Solid Waste Disposal System installed at FNP.

20.302 No such application for proposed disposal procedures, as described in this regulation, has been made or is currently contemplated at APC.

20.303	No plans for waste disposal by release into sanitary sewerage systems, as provided for in this regulation, are contemplated by FNP, nor is this practice currently utilized.
20.304	Disposal of wastes by burial in soil (i.e., onsite burial), as provided for in this regulation, is not performed or being contemplated by FNP.
20.305	Specific authorization, as described in this regulation, is not currently being sought by APC for treatment or disposal of wastes by incineration.
20.401	All of the requirements of this regulation will be complied with through the implementation of appropriate Technical Specifications and health physics procedures pertaining to records of surveys, radiation monitoring and waste disposal. The retention periods specified for such records are also provided for in these specifications and procedures. (See FSAR Chapter 12)
20.402	FNP has established an appropriate inventory and control program to ensure strict accountability for all licensed radioactive materials. Reports of theft or loss of licensed material will be required by reference to the regulations of 10 CFR in the Technical Specifications.
20.403	Notifications of incidents, as described in this regulation will be assured by the requirements of the Technical Specifications, the FNP Health Physics Manual and appropriate plant procedures, which also provide for the necessary assessments to determine the occurrence of such incidents.
20.404	This regulation was deleted effective September 17, 1973 (38 Fed. Reg. 22220).
20.405	Reports of overexposures to radiation and the occurrence of excessive levels and concentrations, as required by this regulation, will be provided for by reference in the Technical Specifications and in appropriate health physics procedures.
20.406	This regulation was deleted August 17, 1973, effective September 17, 1973 (38 Fed. Reg. 22220).
20.407	The personnel monitoring report required by this regulation will be expressly provided for by the Technical Specifications. Appropriate health physics procedures establish the data base from which this report is generated. (See FSAR Chapter 12)

20.408 The report of radiation exposure required by this regulation upon termination of an individual's employment or work assignment is generated through the provisions of the health physics procedure.

20.409 The notification and reporting requirements of this regulation, and those referred to by it, are satisfied by the provisions of the health physics procedure.

20.501 This regulation provides for the granting of exemptions from 10 CFR Part 20 regulations, provided such exemptions are authorized by law and will not result in undue hazard to life or property. It does not impose independent obligations on licensees.

20.502 This regulation describes the means by which the Commission may impose upon any licensee requirements which are in addition to the regulations of Part 20. It does not impose independent obligations on licensees.

20.601 This regulation describes the remedies which the Commission may obtain in order to enforce its regulations, and sets forth those penalties or punishments which may be imposed for violations of its rules. It does not impose any independent obligations on licensees.

Regulation (10 CFR)	Compliance
50.1	This regulation states the purpose of the Part 50 regulations and does not impose any independent obligations on licensees.
50.2	This regulation defines various terms and does not impose independent obligations on licensees.
50.3	This regulation governs the interpretation of the regulations by the NRC and does not impose independent obligations on licensees.
50.4	This regulation gives the address of the NRC and does not impose independent obligations on licensees.
50.10 50.11	These regulations specify the types of activities that may not be undertaken without a license from the NRC. APC does not propose to conduct any such activities at FNP without an NRC license.
50.12	This regulation provides for the granting of exemptions from 10 CFR Part 50 regulations, provided such exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. It does not impose independent obligations on licensees.
50.13	This regulation says that a license applicant need not design against acts of war. It imposes no independent obligations on licensees.
50.20 50.21 50.22 50.23	These regulations merely describe the types of licenses that the NRC issues. They do not address the substantive requirements that an applicant must satisfy to qualify for such licenses.
50.24	This regulation has been deleted, 35 Fed. Reg. 19655.
50.30	This regulation sets down procedural requirements for the filing of license applications, such as the number of copies of the application that must be provided the NRC. APC has substantially complied with the procedural requirements in effect at the time when filing its license application and the amendments to it. In particular, 10 CFR 50.30(f) requires that a license application must be accompanied by any Environmental Report required pursuant to 10 CFR Part 51, and APC has submitted a Final Environmental Statement covering FNP.
50.31 50.32	These regulations merely permit more efficient organization of the license application and impose no independent obligations on licensees.

50.33 This regulation requires the license application to contain certain general information, such as an identification of the applicant, information about the applicant's financial qualifications, and a list of regulatory agencies with jurisdiction over the applicant's rates and services. This information was provided in the FNP operating license application.

50.33(a) This regulation requires applicants for construction permits to submit information required for antitrust review. The antitrust review required by the Atomic Energy Act of 1954, as amended, was performed at the construction permit stage.

50.34(a) This regulation governs the contents of the Preliminary Safety Analysis Report and is relevant to the construction permit stage rather than the operating license stage.

50.34(b) A Final Safety Analysis Report (FSAR) has been prepared and submitted, which addresses in the chapters indicated the information required:

(1) site evaluation factors - Chapter 2

(2) structures, systems, and components - Chapters 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, and 15

(3) radioactive effluents and radiation protection - Chapters 11 and 12

(4) design and performance evaluation - ECCS performance is discussed and shown to meet the requirements of 10 CFR 50.46 in Chapters 6 and 15.

(5) results of research programs - Chapter 1

(6) (i) organizational structure - Chapter 13

(ii) managerial and administrative controls - Chapters 13 and 17. Chapter 17 discusses compliance with the quality assurance requirements of Appendix B.

(iii) plans for preoperational testing and initial operations - Chapter 14

(iv) plans for conduct of normal operations - Chapters 13 and 17. Surveillance and periodic testing is specified in the Technical Specifications.

(v) plans for coping with emergencies - Emergency Plan (Chapter 13)

(vi) Technical Specifications - prepared in conjunction with the Staff (Chapter 16)

(vii) not applicable, since the operating license application was filed before February 5, 1979

(7) technical qualifications - Chapter 13

(8) operator requalification program - Chapter 13 and letters of 6/20/80, 7/15/80, 7/29/80, 8/1/80, and 8/6/80. The latter letters describe the training program including a requalification program meeting the new requirements of NUREG-0694.

50.34(c)	A physical security plan has been prepared and will be implemented.
50.34(d)	A safeguards contingency plan has been prepared and will be implemented.
50.34a 50.35	These regulations are relevant to the construction permit stage rather than the operating stage.
50.36	Technical Specifications have been prepared and will be implemented, including items in each of the categories specified, including: (1) safety limits and limiting safety settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.
50.36a	The Environmental Technical Specifications, Part I, include specifications which require compliance with 10 CFR 50.34a (releases as low as is reasonably achievable), and that ensure that concentrations of radioactive effluents released to unrestricted areas are within the limits specified in 10 CFR 20.106. The reporting requirements of 10 CFR 50.36a(a)(2) are also included in these specifications.
50.37	This regulation requires the applicant to agree to limit access to Restricted Data. APC's agreement to do so is in the operating license application for the Farley Nuclear Plant Units 1 and 2 in Section 11.
50.38	This regulation prohibits the NRC from issuing a license to foreign-controlled entities. APC's statement that it is not owned, controlled, or dominated by an alien, foreign corporation, or foreign government is in the operating license application for the Farley Nuclear Plant.
50.39	This regulation provides that applications and related documents may be made available for public inspection. This imposes no direct obligations on applicants and licensees.

50.40 This regulation provides considerations to "guide" the Commission in granting licenses as follows:

50.40(a) The design and operation of the facility is to provide reasonable assurance that the applicant will comply with NRC regulations, including those in 10 CFR Part 20, and that the health and safety of the public will not be endangered. The basis for APC's assurance that the regulations will be met and the public protected is contained in this enclosure and in the license application and the related correspondence over the years. Moreover, the lengthy process by which the plant is designed, constructed, and reviewed, including reviews by APC's own staff, the NRC staff, the ACRS, and NRC licensing boards, provides a great deal of assurance that the public health and safety will not be endangered. In particular, the Atomic Safety and Licensing Board, after an extensive review, concluded that APC had the commitment and technical qualifications necessary to operate Farley Nuclear Plant safely and in compliance with all applicable radiological health and safety requirements.

50.40(b) Another consideration is that the applicant be technically and financially qualified. Both APC's technical qualifications and its financial qualifications were reviewed in hearings before the Atomic Safety and Licensing Board.

50.40(c) Another consideration is that the issuance of the license is not to be inimical to the common defense and security or to the health and safety of the public. The individual showings of compliance with particular regulations contained in this enclosure, as well as the contents of the entire FSAR and related correspondence over the years, plus the lengthy process of design, construction, and review by APC, its NSSS vendor, and the government, provide APC with considerable assurance that the license will not be inimical to the health and safety of the public. As for the common defense and the security, there is considerable assurance that the license will not be inimical in that APC has an approved security plan for Farley Nuclear Plant that APC is not controlled by agents of foreign countries, and that APC has agreed to limit access to Restricted Data.

50.40(d) The final 50.40 "consideration" is that the applicable requirements of Part 51 have been satisfied. Part 51 concerns compliance with the National Environmental Policy Act of 1969. APC has submitted a Final Environmental Report and the NRC staff has reviewed the report and APC published a final Environmental Impact Statement pursuant to 10 CFR 50, Appendix D. Environmental Technical Specifications are pending.

50.41 This regulation applies to Class 104 licensees, such as those for devices used in medical therapy. Farley Nuclear Plant has not applied for a Class 104 license, and so 50.41 is not applicable.

50.42 Section 50.42 provides additional "considerations" to "guide" the Commission in issuing Class 103 licenses. The two considerations are: (a) that the proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized and (b) that due account will be taken of the antitrust advice provided by the Attorney General under subsection 105c of the Atomic Energy Act. The "useful purpose" to be served is the production of electric power. The need for the power was determined by the licensing board at the construction permit stage. Although conditions affecting the need for power are constantly changing, APC periodically makes load projections, and in APC's judgement the need for Farley Nuclear Plant is still substantial. As for the amount of special nuclear material or source material used, there is no reason to believe that their proportion in relation to the power produced is substantially greater than that of other commercial power reactors in this country. As for the antitrust review, the Commission referred the matter to an Atomic Safety and Licensing Board (Docket Nos. 50-348A and 50-364A) for hearing and determination. Such Board on April 8, 1977, issued its Opinion, Findings and Order in which it was determined that applicant's activities under the Farley Plant license would maintain a situation inconsistent with certain antitrust laws and that it would be necessary to attach additions to the license which will prevent such a result. Subsequently on June 24 such Board at the foot of the further proceeding relating to the matter of what conditions should be attached to the license, issued a further order prescribing definitive license conditions which were appended to the operating license for Farley Unit 1. Exceptions to the determinations made by such Board were filed by applicant, Department of Justice, the Staff of the Nuclear Regulatory Commission, and two intervenors. Such exceptions are now pending before an Atomic Safety and Licensing Appeal Board of the Commission.

50.43 This regulation imposes certain duties on the NRC and addresses the applicability of the Federal Power Act and the right of government agencies to obtain NRC licenses. It imposes no direct obligations on licensees.

50.44 The Farley Nuclear Plant combustible gas control system is described in FSAR Section 6.2.5. The system is designed to maintain the hydrogen concentration in containment at a safe level following a LOCA, without purging the containment atmosphere, as specified in 10 CFR 50.44(e). The system consists of two internal recombiners and two hydrogen analyzers. The containment recirculation system and purge system complement the recombiner system. The Farley Nuclear Plant system meets the requirements of NUREG-0660 and NUREG-0694. The requirements of 10 CFR 50.44 are satisfied.

50.45 This regulation provides standards for construction permits rather than operating licenses and is therefore not material to this operating license proceeding.

50.46 FSAR Sections 6.3 and 15.4.1 describe the Emergency Core Cooling System and the methods used to analyze ECCS performance following a postulated loss of coolant accident.

50.46 (Cont'd)	In FSAR Amendment 72 dated March 7, 1980, APC provided the results of a LOCA-ECCS analysis for Farley Nuclear Plant using an NRC-approved evaluation model which is in compliance with Appendix K to 10 CFR 50. The analysis, based on an overall peaking factor (Fq) of 2.32, provided results in compliance with the criteria of 10 CFR 50.46(b). The Fq limit is reflected in the Technical Specifications.
	The Staff has proposed new standards for fuel cladding swelling and rupture models in draft NUREG-0630. If these standards were to be adopted for the currently approved evaluation models, a decrease in the Fq Technical Specification limit would be required. However, by letter dated August 6, 1980, APC provided an assessment of this reduction as well as the benefits of modeling improvements described to the NRC by Westinghouse Electric Corporation. It was concluded that application of these improvements would more than offset any Fq reduction resulting from the revised staff cladding model; therefore, the existing Technical Specification limit is acceptable.
50.50	This regulation provides that the NRC will issue a license upon determining that the application meets the standards and requirements of the Atomic Energy Act and the regulations and that the necessary notifications to other agencies or bodies have been duly made. It imposes no direct obligations on licensees.
50.51	This regulation specifies the maximum duration of licenses. Compliance will be affected simply by the Commission's writing the license so as to comply.
50.52	This regulation provides for the combining in a single license of a number of activities. It imposes no independent obligation on the licensee.
50.53	This regulation provides that licenses are not to be issued for activities that are not under or within the jurisdiction of the United States. The operation of Farley Nuclear Plant will be within the United States and subject to the jurisdiction of the United States, as is evident from the description of the facility in the operating license application.
50.54	This regulation specifies certain conditions that are incorporated in every license issued. Compliance is effected simply by including these conditions in the license when it is issued. Indeed, much of 50.54 merely provides that other provisions of the law apply, which would be the case even without 50.54.

50.55 This regulation addresses conditions of construction permits, not operating licenses, and so it is not relevant at this point.

50.55a(a)(1) Various chapters of the FSAR discuss design, fabrication, erection, construction, testing, and inspection of safety-related equipment. For example, Chapter 14 provides information on testing of safety-related systems. Chapter 17 provides information concerning the Quality Assurance Program that was utilized. As a further example of a specific system, Chapter 5, Section 5.2, "Integrity of the Reactor Coolant System Boundary," discusses the design of the reactor coolant system.

50.55a(a)(2) This paragraph is a general paragraph leading into paragraphs (c) through (i) of the regulation.

50.55a(b)(1)
50.55a(b)(2) These paragraphs provide guidance concerning the approved Edition and Addenda of Section III and XI of the ASME B&PV Code.

50.55a(c) Design and fabrication of the reactor vessel was carried out in accordance with ASME Section III (1968) Class A. Information can be found in Chapters 3 and 5 of the FSAR.

50.55a(d) Reactor coolant system piping meets the requirements of ASME Section III (1971) Class 1. Information can be found in Chapters 3 and 5 of the FSAR.

50.55a(e) Reactor coolant pumps meet the requirements of ASME Section III (1971). Information can be found in Chapters 3 and 5 of the FSAR.

50.55a(f) The valves within the reactor coolant system pressure boundary were designed and fabricated in accordance with the requirements of ASME Section III, 1971 edition. (See FSAR Chapters 3 and 5)

50.55a(g) Inservice Inspection (ISI) requirements are delineated in this part and are specified in the Technical Specifications, paragraph 4.0.5. As permitted by this part and the Technical Specifications, certain exemptions have been requested for the inservice inspection of various systems and the inservice testing of various pumps and valves. By letters dated March 12, 1980 and July 25, 1980 the Farley Nuclear Plant-Unit 2 Inservice Testing Program for Pumps and Valves and Inservice Inspection Program were docketed. The Preservice Inspection Program and relief requests for this program are incorporated in FSAR Section 5.2.8.

50.55a(h) As discussed in Chapter 7, the protection systems meet IEEE 279-1971.

50.55a(i) Fracture toughness requirements are set forth in Appendices G and H of 10 CFR 50. Technical Specifications require the use of reactor vessel material irradiation surveillance specimens and updating of the "heatup" and "cooldown" curves given in the Technical Specifications. Further information on this subject is given in Chapter 5 of the FSAR and in APC letter to the NRC dated July 30, 1980.

50.55b This regulation has been revoked. 43 Fed. Reg. 49775.

50.55e This regulation is only proposed, 39 Fed. Reg. 26297, and applies to fuel reprocessing plants.

50.56 This regulation provides that the Commission will, in the absence of good cause shown to the contrary, issue an operating license upon completion of the construction of a facility in compliance with the terms and conditions of the construction permit. This imposes no independent obligations on the applicant.

50.57(a) This regulation requires the Commission to make certain findings before the issuance of an operating license. Specifically, the findings are:

(1) Construction of the facility has been substantially completed in conformity with the construction permit and the application as amended. Conformance of the facility to the NRC rules and regulations and the Act, as implemented by the regulations, has been demonstrated by the application.

(2) The Technical Specifications and resulting operating procedures provide assurance that the unit will operate in conformity with the application as amended and with the rules and regulations.

(3) The application demonstrates that the facility can be operated without endangering the health and safety of the public and in compliance with the regulations, as noted above.

(4) The application demonstrates that APC is technically and financially qualified to operate the unit.

(5) The applicable provisions of 10 CFR 140 have been satisfied.

(6) The approved Security Plan assures that special nuclear material is being appropriately safeguarded. The application demonstrates that the operation of the unit will not be inimical to the health and safety of the public.

50.57(b) The license, as issued, will contain appropriate conditions to assure that items of construction or modification are completed on a schedule acceptable to the Commission.

50.57(c) This regulation provides for a low-power testing license.

50.58 This regulation provides for the review and report of the Advisory Committee on Reactor Safeguards. The ACRS has reviewed the operating license application for FNP in accordance with its usual practice.

50.59 This regulation provides for the licensing of certain changes, tests, and experiments at a licensed facility. Technical Specifications and procedures provide implementation of this regulation.

50.60 This regulation has been deleted, 40 Fed. Reg. 8790.

50.65 This regulation has been deleted, 43 Fed. Reg. 6915.

50.70 The Commission has assigned resident inspectors to the FNP. APC has provided office space in accordance with the requirements of this section. APC permits access to the station to NRC inspectors in accordance with 10 CFR 50.70(b)(3).

50.71 Records are and will be maintained in accordance with the requirements of sections (a) through (e) of this regulation and the license. Section (e) requires that the FSAR be updated by July 22, 1982, and annually thereafter. Such updates will be made.

50.72 Notification of significant events to the NRC will be made in accordance with the requirements in this regulation.

50.78 This regulation governs the contents of the Preliminary Safety Analysis Report and is relevant to the construction permit stage rather than the operating license stage.

50.80 This regulation provides that licenses may not be transferred without NRC consent. No application for transfer of a license is involved in the FNP proceeding.

50.81 This regulation permits the creation of mortgages, pledges, and liens on licensed facilities, subject to certain provisions. APC licensed facilities are not mortgaged, pledged, or otherwise encumbered as those terms are used in this regulation.

50.82 This regulation provides for the termination of licenses. It does not apply to FNP because APC has not requested the termination of a license.

50.90 This regulation governs applications for amendments to licenses. Future requests for license amendments will be made in accordance with these requirements.

50.91 This regulation provides guidance to the NRC in issuing license amendments.

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These regulations govern the revocation, suspension, and modification of licenses by the Commission under unusual circumstances. No such circumstances are present in the FNP proceeding, and these regulations are not applicable.

50.109

This regulation specifies the conditions under which the NRC may require the backfitting of a facility. This regulation imposes no independent obligations on a license unless the NRC proposes a backfitting requirement, and so this regulation is not applicable.

50.110

This regulation governs enforcement of the Atomic Energy Act, the Energy Reorganization Act of 1974, and the NRC's regulations and orders. No enforcement action is at issue in the FNP proceeding, and so this regulation is not applicable.

Appendix A

GDC 1

Section 3.1.1 of the FSAR describes the design provisions made to ensure that these requirements are met. Codes and standards utilized for the unit are specified throughout the FSAR. Chapter 17 describes the quality assurance program and the provisions for maintenance of records.

GDC 2

FSAR Section 3.1.2 addresses the design considerations for natural phenomena, which are described in detail in Chapters 2 and 3. Appropriate considerations have been made in the design basis for historical data, combined effects of normal and accident conditions with the effects of natural phenomena, and the importance of the safety functions to be performed.

GDC 3

FSAR Section 3.1.3 describes in general the measures which have been taken to minimize the probability and effects of fires and explosions. Section 9.5.1 describes the fire detection and protection systems. In addition, improvements to the fire protection systems have been and are being made in accordance with NRC requirements based on Appendix A to BTP APCSB 9.5-1. These modifications will be completed by November 1, 1980. This schedule is in accordance with the Commissioner's Order dated May 23, 1980

GDC 4 Structures, systems, and components important to safety are designed to accommodate the effects of any environmental conditions at the Farley site that are associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and to be compatible with these conditions. Design criteria and implementation are presented in Sections 3.5 and 3.6, and environmental factors are described in Sections 3.11, 6.3, and 6.4. These structures, systems, and components are appropriate protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from any postulated failure of equipment or structural design or from environmental conditions and accidents outside the nuclear power unit.

Chapter 7.0 lists the motors, instrumentation, and associated cables of protection and safety features systems located inside the containment structure. It also gives the design requirements in terms of the time that each must survive the extreme environmental conditions following a LOCA.

Details of the design, environmental testing, and construction of these systems, structures, and components are included in other sections of Chapters 3.0, 5.0, 6.0, 7.0, and 9.0. Evaluation of the performance of safety features and analyses of possible accidents are contained in Chapter 15.0.

In addition, a review of scoped class 1E electrical equipment inside and outside the containment against NUREG-0588 is near completion. This review has confirmed that most electrical equipment has been adequately demonstrated to be qualified. The results of this review are scheduled to be submitted to the staff by September 15, 1980.

GDC 5	As described in FSAR Section 3.1.5, those structures, systems and components which are shared with Unit 1 are tabulated in FSAR Section 1.2.12. It is concluded that safety functions are not significantly impaired by such sharing.
GDC 10	FSAR Section 3.1.6 indicates that the reactor core and associated systems are designed to function throughout the design lifetime without exceeding fuel damage limits, using protection criteria specified in Section 3.1.6, and Chapters 4, 7, and 15.
GDC 11	FSAR Section 3.1.7 indicates that prompt compensatory reactivity feedback effects are assured by unit design and operational limit considerations. The core inherent reactivity feedback characteristics and reactivity control methods are described in FSAR Sections 4.3, 4.2.3, and 9.3.4.
GDC 12	FSAR Section 3.1.8 describes the inherent and design features which eliminate or limit the various types of oscillations. Core stability is further described in Section 4.3 and Chapter 7.
GDC 13	As indicated in FSAR Section 3.1.9, and described in more detail in Chapters 4 and 7, instrumentation and control systems have been provided to monitor and maintain plant variables including those variables which affect the fission process, integrity of the reactor core, the reactor coolant pressure boundary, and the containment, over their prescribed ranges for normal operation, anticipated occurrences, and under accident conditions.
GDC 14	FSAR Section 3.1.10 indicates that the reactor coolant pressure boundary has been designed to accommodate the system temperatures and pressures attained under all expected operational modes and anticipated transients, and to maintain stresses within applicable limits. Design criteria and methods are described in Sections 3.1.10, 5.2.1, 3.6, 3.7, 5.2.2, 5.2.4, 5.2.5, 5.2.7, and 5.2.8.
GDC 15	As indicated in FSAR Section 3.1.11, the reactor coolant system and associated auxiliary, control and protection systems are designed to ensure the integrity of the reactor coolant pressure boundary with adequate margins during normal operations and anticipated transients. The design codes used for the Reactor Coolant System are described in Chapter 5. Details concerning the protection systems are provided in Section 7.2.2.
GDC 16	As described in FSAR Sections 3.1.12 and 3.8.1, and Chapter 6, a reinforced concrete, steel-lined containment structure is provided. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures, up to and including the rupture of the largest pipe in the Reactor Coolant System. The containment and its associated Engineered Safety Features thus meet the required functional capability of this criteria.

GDC 17 As described in FSAR Section 3.1.13, onsite and offsite power systems are provided which can independently supply the electric power required for the operation of the safety related systems. This capability is maintained even with the failure of any single active component in either system. Chapter 8 provides the design details of the power systems and their compliance with this criterion.

GDC 18 As described in FSAR Sections 3.1.14, 8.1.4, 8.3.1, and 8.3.2, the redundant electric power systems important to safety are continuously monitored and energized during normal plant operation from redundant offsite power sources. Redundant offsite diesel generators provide automatic backup power sources. Periodic tests of the diesel generators, the transfer system and the station batteries are made, as required by Technical Specifications.

GDC 19 FSAR Section 3.1.15 describes the Main Control Room, which contains the controls and instrumentation necessary for safe operation of the unit during normal and accident conditions. Sufficient shielding, distance, structural integrity, and ventilation systems are provided to ensure that control room personnel will not receive radiation exposures in excess of the criterion for the duration of the accident. In the event that access to the main control room is restricted, auxiliary shutdown panels have been provided, within the protected envelope, which may be used to maintain the reactor in a hot shutdown condition. Subsequent cold shutdown may be achieved using suitable procedures.

GDC 20 FSAR Section 3.1.16 discusses the design criteria for the protection system and engineered safety features actuation, to ensure that the requirements of this criterion are met. Further details are supplied in Chapter 7.

GDC 21 As indicated in FSAR Section 3.1.17, the protection system is designed for the high functional reliability and inservice testability commensurate with the safety functions to be performed. This section, as well as Chapter 7, describes in detail the design features provided to ensure redundancy and testability.

GDC 22 FSAR Section 3.1.18 indicates that the protection system has been designed to provide sufficient resistance to a broad class of accident conditions or postulated events. Chapter 7 provides further design details concerning this resistance such that independence is maintained.

GDC 23 As indicated in FSAR Section 3.1.19, the protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of energy sources and the environment. Further details are supplied in Chapter 7.

GDC 24	FSAR Section 3.1.20 discusses separation of the protection and control systems, such that the failure of any single control system component or channel or the failure or removal from service of any protection system component or channel which is common to the protection and control systems, leaves intact a system satisfying all redundancy, reliability, and independence requirements of the protection system. Details concerning separation of protection and control systems are provided in Chapter 7.
GDC 25	FSAR Section 3.1.21 indicates that the protection system has been designed to assure that specified acceptable fuel design limits are not exceeded in the event of any single reactivity control system malfunction, including an accidental withdrawal of control cluster groups. Further details are provided in FSAR Sections 4.3.1.4, 4.2.3, 7.2 and 7.7.
GDC 26	As indicated in FSAR Section 3.1.22, two independent reactivity control systems of different design principles are provided. One of the systems uses control rods; the second system employs dissolved boron. Reactivity control system redundancy and capability are described further in Sections 4.3.1.5, 4.2.3, and 7.7.
GDC 27	As described in FSAR Section 3.1.23, means are provided for shutdown reactivity for cooling the core under any anticipated condition and with appropriate margin for contingencies. Combined use of rod cluster control and chemical shim control permit the necessary shutdown margin to be maintained during the long term xenon decay and plant cooldown. These means are discussed in detail in FSAR Sections 4.2.3 and 9.3.4.
GDC 28	FSAR Section 3.1.24 indicates that core reactivity is controlled by a chemical poison dissolved in the coolant, rod cluster assemblies and burnable poison rods. The maximum reactivity insertion rates due to withdrawal of a bank of rod cluster control assemblies or by boron dilution are limited. The maximum worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values which prevent rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity. Further details are provided in Sections 4.2.3 and 4.3.
GDC 29	As indicated in FSAR Section 3.1.25, the protection and reactivity control systems are designed to assure extremely high probability of performing their required safety functions in the event of anticipated operational occurrences. The protection system is further discussed in Section 7.2. The reactivity control systems are discussed in Sections 4.2 and 7.7.

GDC 30 As described in FSAR Section 3.1.26, reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with ASME Nuclear Power Plant Components Code Section III. Major components are classified as seismic Class 1 and are accorded the quality measures appropriate to this classification. The evaluations of reactor coolant pressure boundary components are discussed in Section 5.2.

GDC 31 As indicated in FSAR Section 3.1.27 and 5.2.4, close control is maintained over material selection and fabrication for the Reactor Coolant System to assure that the boundary behaves in a nonbrittle manner. The materials testing is consistent with 10 CFR 50, Appendices G and H. These tests ensure the selection of materials with proper toughness properties and margins as well as verify the integrity of the reactor coolant pressure boundary. Operating procedures and Technical Specifications ensure operation within the pressure-temperature limit relative to this criterion.

GDC 32 FSAR Section 3.1.28 indicates that the design of the reactor coolant pressure boundary provides the capability for accessibility. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. Additional details can be found in Sections 5.2.4 and 5.2.8.

GDC 33 As indicated in FSAR Section 3.1.29, the Chemical and Volume Control System provides a means of reactor coolant makeup and adjustment of the boric acid concentration. A high degree of functional reliability and safe response to probable modes of failure is assured by provision of standby components. Details of system design are included in Section 9.3.4 and details of the electrical power systems are given in Sections 8.2 and 8.3.

GDC 34 FSAR Section 3.1.30 indicates that the Residual Heat Removal System, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core within acceptable limits. Suitable redundancy is accomplished below 350°F with the two residual heat removal pumps with means available for draining and monitoring of leakage, two heat exchangers, and the associated piping and cabling. The Residual Heat Removal System is able to operate on either onsite or offsite electrical power. Suitable redundancy above 350°F is provided by the steam generators, auxiliary feed pumps, and attendant piping. Details of the Residual Heat Removal System design are in FSAR Sections 5.5.7 and 6.3.

GDC 35 FSAR Section 3.1.31 describes the use of passive accumulators with two centrifugal charging pumps and two low head safety injection pumps to provide redundancy for failure of any component in any system. The primary function of the Emergency Core Cooling System is to deliver borated cooling water to the reactor core in the event of a loss-of-coolant-accident. This limits the fuel clad temperature and thereby ensures that the core will remain substantially intact and in place, with its essential heat transfer geometry preserved. Further details are provided in Section 6.3.

GDC 36 As described in FSAR Section 3.1.32, design provisions are made for inspection to the extent practical of all components of the Emergency Core Cooling System. An inspection is performed periodically to demonstrate system readiness. To the extent possible, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence. Non-destructive inspection is performed where such techniques are desirable and appropriate. Technical Specifications require inservice inspection in accordance with applicable ASME Codes. Details of the inspection programs are provided in Sections 5.4 and 6.3.4.

GDC 37 FSAR Section 3.1.33 indicates that the components of the Emergency Core Cooling System located outside the containment will be accessible for leak-tightness inspection during appropriate periodic tests. Each active component of the system may be individually actuated on the normal power source at any time during plant operation to demonstrate operability. The centrifugal charging pumps are part of the charging system, and this system is in continuous operation during plant operation. Actuation circuits are tested and remote operated valves are exercised periodically. The testing is described in detail in FSAR Section 6.3.4 and as per Technical Specification surveillance requirements.

GDC 38 The containment cooling system, consisting of four fan coolers, has the capacity to remove the full containment heat load during an accident condition. Even though four coolers will be automatically started at low speed upon receipt of a safety injection signal, only two are required for 100 percent design heat removal capability. Reduction in the heat removal capacity caused by failure of a single active component in either system will be compensated for by the other system. The containment spray system also lowers containment atmosphere temperature and pressure during the injection phase.

The containment heat removal systems are designed to ensure that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the systems from accomplishing their design safety functions. For a more complete description of the design and operation of the containment heat removal systems, see Sections 6.2.2 and 3.1.34.

GDC 39 The containment air coolers and associated service water system piping (Section 6.2.2) inside containment can be inspected during shutdown. The service water system is outside containment, and, with the exception of buried piping, can also be inspected periodically.

The containment spray system (Section 6.2.2) essential equipment except risers, distribution header piping, spray nozzles, and the containment sump, are located outside containment. The containment sump and the spray pipe and nozzles can be inspected during shutdowns. Those portions of the containment spray suction piping from the containment sump and refueling water storage tank either embedded in concrete or buried are not accessible for inspections. Associated equipment outside the containment can be visually inspected. (See FSAR Section 3.1.35)

GDC 40 The normal operation of the four containment coolers will verify structural and leaktight integrity of the system components, as well as demonstrate the operation of the associated cooling system. System piping, valving, pumps, fans (at low speed), coolers and other components of this system are arranged so that each component can be tested periodically for operability (Section 6.2.2).

The delivery capability of the containment spray system can be tested periodically to the extent practical up to the last valve before the spray nozzles. These tests can also verify the structural and leak-tight integrity of the spray system components (Section 6.2.2).

The containment air cooling and containment spray systems are designed to provide the capability for testing the full operational sequence as close to design as practical, including transfer to alternate power sources, during shutdown or refueling (Sections 6.2.2.4 and 6.2.3.4. Also see FSAR Section 3.1.36)

GDC 41 Sodium hydroxide additive to the containment spray solution is provided to reduce airborne iodine activity levels inside containment and to retain the removed iodine in solution in the core sump. For a complete description of the design of the containment spray system, including spray additive capability, see Sections 6 2.2 and 6.2.3.

A penetration room filtration system is provided for control of contaminants which might leak from the containment through the penetrations following a LOCA. This system is also used to mitigate the consequences of a fuel handling accident. For a complete description of the design of the penetration room filtration system, see Section 6.2.3.

A hydrogen control system is provided to control hydrogen concentrations inside containment following an accident. This system consists of redundant hydrogen-oxygen recombiners, and appropriate containment atmosphere mixing and sampling systems. Backup protection against excessive containment

GDC 41 (Cont'd) atmosphere hydrogen concentrations is provided by a post-accident containment purging system. For a complete description of the design of the containment combustible gas control system, see Section 6.2.5.

Each of the containment atmosphere cleanup systems has suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure. (See FSAR Section 3.1.37)

GDC 42 FSAR Section 3.1.38 indicates that the Containment Atmosphere Cleanup System and the Containment Depressurization System are designed to permit appropriate periodic inspection of the important components. Additional discussion is provided in FSAR Section 6.2.

GDC 43 The capability for testing of the containment spray system has been discussed in the response to Criterion 40.

The penetration room filtration system is designed to provide the capability for testing the penetration room filtration system through the full operational sequence and as close to design conditions as practical, including transfer to alternate power sources, during shutdown or refueling. Such tests would assure the operability and performance of the active system components, and would demonstrate the structural and leaktight integrity of the system components. A capability is also provided for periodic testing and surveillance of the penetration room filtration system filters to ensure that (a) filter bypass paths have not developed and (b) filter and adsorber materials have not deteriorated beyond acceptable limits.

The post-accident containment combustible gas control system is designed to allow appropriate periodic testing of the system through the full operational sequence and as close to design conditions as practical, including transfer to alternate power sources, to demonstrate system integrity as well as operability and performance of system active components. For further discussion of the testing capability of this system see FSAR Section 6.2.5. (See FSAR Section 3.1.39)

GDC 44 Component cooling water, service water, river water, and spent fuel pool cooling systems are provided to transfer heat from the plant to an ultimate heat sink. These systems have been designed to transfer their respective heat loads under all anticipated operating and accident conditions. Suitable redundancy, leak detection, and system interconnection and isolation capabilities have been incorporated into the design of these systems to assure that the systems can accomplish all required safety functions, assuming a single failure concurrent with either onsite or offsite power exclusively.

GDC 44 (Cont'd) — A more complete description of the spent fuel pool cooling system design is presented in Section 9.1.3. Descriptions of the design of the other cooling water systems are presented in Section 9.2. (See FSAR Section 3.1.40)

GDC 45 — All safety-related components of the cooling water systems, with the exceptions of buried piping to the river and to pond intake structures and service water piping to the containment coolers, that are located within the containment, are accessible for periodic inspection. The service water piping inside the containment is accessible for inspection during reactor shutdown and refueling periods. The integrity of the buried pipe is demonstrated by pressure and functional tests. A more detailed discussion is contained in Section 9.2. (See FSAR Section 3.1.41)

GDC 46 — All cooling water systems operate either continuously or intermittently during normal plant operation. This functional operation serves to demonstrate the operability, performance, and structural and leaktight integrity of the system components. The cooling water systems are designed to include the capability for testing through the full operational sequence, as close to accident design conditions as practical, including transfer to alternate power sources. A more detailed discussion is contained in FSAR Section 9.2. (See FSAR Section 3.1.42)

GDC 50 — As discussed in FSAR Section 3.1.43, the design of the containment is based on the loss of coolant accident (LOCA), coupled with the partial loss of the redundant engineered safety features, which produces the post-LOCA temperature and pressure conditions described in Section 6.2.1.

A minimum safety margin of 10 percent between the peak calculated post-LOCA containment pressure and the design pressure has been provided.

The containment structural design is described in Section 3.8.

GDC 51 — As discussed in FSAR Section 2.1.44, the design condition of the containment pressure boundary is based on the parameters derived after the design basis accident. For this design condition, the steel liner material behaves in a nonbrittle manner and has the capability to minimize the propagation of any undetected flaw.

Principal load-carrying components of ferritic materials used in the containment will not be exposed to the external environment. The ferritic material of the containment liner plate is designed to function as a leaktight membrane only.

In addition, Nil Ductility Transition Temperature (NDTT) requirements are not considered relevant for the design of the containment since this is a ligament type of structure wherein the brittle fracture of a ligament could not propagate to adjacent ligaments.

GDC 51 (Cont'd)

In all areas where the liner plate is the pressure resisting structural element without backup from the concrete, a minimum NDTT of 0 F has been specified, based on a minimum service temperature of 30 F. These areas are the containment access openings which are enclosed on the outside by heated buildings. The equipment hatch not protected by the auxiliary building may experience service temperatures as low as 0 F; this material has been specified to an NDTT of -30 F.

In all other areas, the liner plate has no structural function since it is backed up by concrete. Except for small locally thickened areas (up to 2 in.) in the floor and walls for crane brackets, anchorages for main steam pipe ruptures, frames, and similar items, the containment liner plate is 1/4-in. thick. No NDTT has been specified since the rules for NDTT from Section III of the ASME Code for Class B vessels apply to pressure vessels only.

See FSAR Section 3.8 for details.

GDC 52

As indicated in FSAR Section 3.1.45, the containment design permits overpressure strength testing during construction and permits preoperational integrated leakage rate testing at containment design pressure and at reduced pressure in accordance with Appendix J, 10 CFR 50. The containment and other equipment which may be subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. However, reduced pressure tests are planned for the surveillance program, in accordance to the Type A test guidelines in Appendix J of 10 CFR 50. The preoperational integrated leak tests at peak pressure and at reduced pressure verify that the containment, including the isolation valvew and the resilient penetration seals, leaks less than the allowable value. Details concerning the conduct of periodic integrated leakage rate tests are in Section 6.2.1.4.

GDC 53

FSAR Section 3.1.46 indicates that the containment and the containment isolation system (Section 6.2) are designed so that: (1) integrated leak tests can be run during plant lifetime (see compliance to Criterion 52), (2) visual inspections can be made of all important areas, such as penetrations, (3) an appropriate surveillance program can be maintained (Section 6.2), (4) periodic testing at containment design pressure of the leaktightness of isolation valves and penetrations which have resilient seals and expansion bellows is possible, and (5) the operability of the containment isolation system can be demonstrated periodically. Further discussion is provided in FSAR Sections 3.8.1.7 and 6.2.1.

GDC 54

As described in FSAR Section 3.1.47, the containment isolation features are classified as Seismic Category I. These components require quality assurance measures which enhance reliability. The containment isolation design provides for a double barrier at the containment penetration in those fluid systems that are not required to function following a design basis event. All

GDC 54 (Cont'd) piping systems penetrating the containment, in so far as practical, have been provided with tests vents and test connections or have other provisions to allow periodic leak testing as required. Section 6.2.1.4 has further details on testing. See Section 6.2.4 for general containment isolation details.

GDC 55 As discussed in FSAR Section 3.1.48, each line that is part of the reactor coolant pressure boundary and that penetrates the reactor containment is provided with containment isolation valves, in accordance with this criterion, as described in FSAR Section 6.2.4.

Isolation valves outside the containment are located as close to the containment as practical, and automatic isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them are provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, includes consideration of the population density, use characteristics and physical characteristics of the site environs.

GDC 56 As discussed in FSAR Section 3.1.49, each line that connects directly to the containment atmosphere and penetrates the primary reactor containment is provided with containment isolation valves in accordance with this criterion, as discussed in Section 6.2.4. Isolation valves outside the containment are located as close to the containment as practical. Upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

GDC 57 As described in FSAR Section 3.1.50, each line that penetrates containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere is provided with an appropriate containment isolation valve arrangement. Refer to Section 6.2.4 for a complete description of the design and operation of the containment isolation system.

GDC 60 As described in FSAR Section 3.1.51, control of waste gas effluents is accomplished by holdup of waste gases in decay tanks until the activity of tank contents and existing environmental conditions permit discharges within 10 CFR 20 and 10 CFR 50 requirements. Waste gas effluents are monitored at the point of discharge for radioactivity and rate of flow. Sufficient waste gas holdup capacity is provided as discussed in Section 11.3, to cope with all anticipated operational occurrences and site environmental conditions. A decay tank burst would not result in an activity release greater than 10 CFR 100 limits, based on one percent failed fuel.

GDC 60 (Cont'd) Control of liquid waste effluents is accomplished by holdup of waste liquids in storage tanks, batch processing of all liquids and sampling before controlled rate discharge. Liquid effluents are monitored for radioactivity and rate of flow. The liquid waste disposal system tankage and evaporator capacity, as described in Section 11.2, is sufficient to cope with all anticipated operational occurrences and unfavorable site environmental conditions.

Station solid wastes will be prepared in batches for offsite disposal by approved contractors in shielded and reinforced containers which meet Federal Regulation requirements. Sufficient handling capacity is provided, as discussed in Section 11.5, to cope with all anticipated operational occurrences.

Chapters 11.0 and 15.0 provide additional information.

GDC 61 All fuel storage and waste handling facilities are contained in the auxiliary building; equipment is designed to prevent accidental releases of radioactivity directly to the environment. Components of these systems which are important to safety can be periodically inspected and tested.

The spent fuel storage pool is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor. Unit 1 is designed to accommodate a total of 4-1/4 cores and Unit 2 a total of 4-1/4 cores. The shipping cask is accommodated in a separate pool adjoining the spent fuel pool.

The spent fuel storage racks are located to provide sufficient shielding water over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mr/hr during the storage period. The exposure time during refueling will be limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The waste disposal system is designed to permit controlled handling and disposal of liquid, gaseous, and solid wastes generated during the plant operation. The principal design criterion is to ensure that plant personnel and the general public are protected against exposure to radiation from wastes in accordance with limits defined in 10 CFR 20. (See FSAR Section 3.1.52)

GDC 62 Criticality in new and spent fuel storage areas is prevented by physical separation of fuel assemblies. In addition, the presence of borated water controls the criticality in the spent fuel storage pool. The separation is provided so that criticality is precluded even if pure water replaces the air gap in the new fuel storage or the borated water in the spent fuel storage area. Criticality prevention is discussed in Section 9.1. (See FSAR Section 3.1.53)

GDC 63 Monitoring systems are provided to alarm on excessive temperature or low water level in the spent fuel pool. In the event of a high temperature alarm, administrative procedures will provide for checking the cooling water flow to the fuel pool coolers, the operating status of the fuel pool cooling pumps, and the integrity of the fuel pool cooling and purification system. In the event of high airborne gaseous or particulate activity, automatic diversion of the normal full building or waste disposal building ventilation system exhaust to the supplementary leak collection and release system will occur.

A radiation monitor and an alarm are provided, as required, to warn personnel of an increase in the level of radiation or airborne activity. The radiation monitoring system is described in Chapters 11.0 and 12.0. (See FSAR Section 3.1.54)

GDC 64 This criterion requires means be provided for monitoring radioactive releases. FSAR Section 3.1.55 indicates that monitoring is provided for the containment atmosphere, effluent discharge paths, and the safeguards areas. Further details are provided in Chapters 11 and 12.

Appendix B Chapter 17 of the FSAR describes in detail the provisions of the quality assurance program which has been implemented to meet all applicable requirements of Appendix B.

Appendix C This Appendix provides a guide for establishing the applicant's financial qualification. APC's financial qualifications were fully litigated before the Atomic Safety and Licensing Board, and the Board expressly found that APC had satisfied the burden of proving that it has reasonable assurance of having the funds that it needs to operate the facility in compliance with the Commission's regulations.

Appendix D This Appendix has been superseded by 10 CFR Part 51. As noted in the discussion for 10 CFR 50.40(d), the requirements of Part 51 have been satisfied.

Appendix E This Appendix specifies requirements for emergency plans. An emergency plan was prepared for FNP before the granting of an operating license for unit 1. This Emergency Plan was litigated before the Atomic Safety and Licensing Board, and the Board found that the emergency plan provided reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect public health and safety and prevent damage to property.

In response to new criteria for emergency planning developed subsequent to the event at Three Mile Island unit 2, the emergency plan has been extensively modified and improved.

Appendix E (Cont'd)	This revised plan, which meets the criteria in NUREG-0654 has been submitted to the NRC staff.
Appendix F	This Appendix applies to fuel reprocessing plants and related waste management facilities, not to power reactors and is therefore not applicable to this proceeding.
Appendix G	Fracture toughness requirements of this Appendix and program requirements given in Appendix H form the basis for Technical Specification surveillance requirements dealing with the use of surveillance specimens. Additional information to demonstrate compliance can be found in FSAR Chapter 5 and APC letter to the NRC dated July 30, 1980. Heatup and cooldown limits consistent with the requirements of this Appendix are established in the Technical Specifications.
Appendix H	Reactor vessel material surveillance program requirements are delineated in this part. Technical Specification 4.4.10.1.2 and operating procedures have been established to implement these requirements with a further requirement to update the "heatup" and "cooldown" curves provided in the Technical Specifications. Further information is provided in FSAR Chapter 5 and APC letter to the NRC dated July 30, 1980.
Appendix I	This Appendix provides numerical guides for design objectives and limiting conditions for operation to meet the criteria "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents. APC has filed with the Commission the necessary information to permit an evaluation of the FNP with respect to the requirements of Section II.A., II.B, and II.C of Appendix I. In this submittal APC provided the necessary information to show conformance with the Commission's September 4, 1975 amendment to Appendix I rather than perform a detailed cost-benefit analysis required by Section II.D of Appendix I.
Appendix J	Reactor containment leakage testing for water cooled power reactors is delineated in this Appendix. These requirements are given in Technical Specification 3/4.6.1. Additional information concerning compliance can be found in FSAR Chapters 3 and 6.
Appendix K	This Appendix specifies features of acceptable ECCS evaluation models. As noted above [illegible] 46, the analysis for FNP has been conducted using [illegible] which has been accepted by the Commission staff as meeting the requirements of this Appendix.
Appendix L	The antitrust review relating to the license applications is now pending before an Atomic Safety and Licensing Appeal Board.

Appendix M	This Appendix covers standardization of design and is not applicable to FNP.
Appendix N	This Appendix covers standardization of nuclear power plant designs and is not applicable to FNP.
Appendix O	This Appendix covers standardization of design and is not applicable to FNP.
Appendix P	This Appendix is proposed, 39 Fed. Reg. 26293, and it applies to fuel reprocessing plants. Accordingly, it is not applicable to FNP.
Appendix Q	This Appendix governs preapplication early review of site suitability issues and is not applicable to FNP.
Appendix Q (Proposed)	This Appendix is proposed, 39 Fed. Reg. 26297, and it would apply to fuel reprocessing plants, not power reactors.

Regulation (10 CFR)	Compliance
100.1	This regulation is explanatory and does not impose independent obligations on licensees.
100.2	This regulation is explanatory. FNP is not novel in design and is not unproven as a prototype or pilot plant.
100.3	This regulation is explanatory and does not impose independent obligations on licensees.
100.10	The factors listed related to both the unit design and the site have been provided in the application. Site specifics, including seismology, meteorology, geology, and hydrology, are presented in Chapter 2 of the FSAR. The exclusion area, low population zone, and population center distance are provided and described. The FSAR also describes the characteristics of reactor design and operation.
100.11	An exclusion area has been established, as described in FSAR Section 2.1.2.1. The low population zone required by 100.11 (a)(2) has been established, as described in FSAR Section 2.1.3.3, as the area within a radial distance of two (2) miles from the centerline of Unit No. 1 containment. As indicated in Section 2.1.3.5, the nearest population center, as defined by 10 CFR 100.3(c), based on the 1975 census, is Dothan, Alabama, which is 16.5 miles west of the site. The FSAR accident analyses, particularly those in Chapters 6 and 15, demonstrate that offsite doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.
Appendix A	Appendix A to 10 CFR 100 provides seismic and geologic siting criteria for nuclear power plants. The compliance of the FNP site with this appendix is discussed in Sections 2.5, 3.2, 3.7, 3.8, and 3.10 in the Farley Nuclear Plant Safety Evaluation Report, NUREG-75/034.