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September 28, 2007

Office of the Secretary
U.S. Nuclear Regulatory Commission,
Washington, D.C. @05555-001
Attention: Rulemaking and Adjudications Staff

RE: In the matter of ENTERGY NUCLEAR INDIAN POINT 2, L.L.C. and ENTERGY NUCLEAR OPERATIONS, INC, Indian point License Renewal Application No. DPR-26, Docket No. 50-247

Dear Sir or Madam:

Please find enclosed for filing in the above stated matter, Friends United for Sustainable Energy USA Inc.'s corrected petition for Leave to Intervene, Request and Contentions; and Notice of Appearance by Attorney Susan H. Shapiro, and finally notice of appearance declarations and exhibits.

Additionally enclosed is a CD ROM digital copy for inclusion into the hearing docket files as well as ADAMS. The original is enclosed together with two copies for your office. In addition a copy is was sent directly to Mr, Larry Burns, Esq., Chief Counsel to the Commission. As he suggested in our meeting with him last Friday, rather that leaving the original and your two copies at the security desk, you would be best served if we over-night the original and two copies to you direcltly. Given my travel plans over the weekend, I was only able to transmit these to you today.

Thank you for your attention in this matter.

Sincerely,

Susan Shapiro, Esq.

Enclosures:
cc: Certificate of Service.

Template Secy-037

Secy 07

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, LLC	)	
And ENTERGY NUCLEAR OPERATIONS, INC.	)	NO. 50-247
Indian Point Energy Center Unit 2	)	

NOTICE OF APPEARANCE

Through Attorney, Susan H. Shapiro, and pursuant of 10 CFR. § 2.314(b) gives notices of her appearance on behalf of Friends United for Sustainable Energy USA, Inc. The undersigned is a member of good standing of the bar of one or more Courts of the United States, and have been duly retained by Friends United for Sustainable Energy to represent it in this matter.. Friends United for Sustainable Energy's address is 21 Perlman Drive, Spring Valley, NY 10977. It's email address is fuseusa@yahoo.com.

By: _____

Susan H. Shapiro
Attorney for Friends United for
Sustainable Energy USA, Inc.
21 Perlman Drive
Spring Valley, NY 10977
(845) 371-2100
mbs@ourrocklandoffice.com

CERTIFICATE OF SERVICE

I hereby certify that on this 19th day of September, 2007, a copy of Friends United for Sustainable Energy USA, Inc.'s Petition for Leave to Intervene, Request for Hearing, and Contention regarding the matter of Entergy Indian Point 2, LLC and Entergy Nuclear Operations, Inc, Indian Point 2 LLC License Renewal Application, Docket No. 50--247, License No. DPR-26 ; and the Notice of Appearance for Friends United for Sustainable Energy USA, Inc. by Attorney Susan H. Shapiro, were sent by First Class U.S. Mail, postage prepaid to:

Office of the Secretary of Commission Congresswoman Nita Lowey
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Att: Rulemaking and Adjudications Staff
(also by e-mail)

Office of the General Counsel Congressman Maurice Hinchey
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
(also by e-mail)

Governor Eliot Spitzer
Eliot Spitzer
State Capitol
Albany, NY 12224

Attorney General Andrew Cuomo
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The Honorable Hillary Rodham Clinton
United States Senate
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Senator Charles Schumer
The Honorable Charles Schumer
United States Senate
313 Hart Senate Office Building
Washington, D.C. 20510-3201

Congressman John Hall
United States House of Representatives
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Congresswoman Nita M. Lowey
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DOCKETED
USNRC

October 25, 2007 (3:00pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C	)	License No. DPR 26
And Entergy Nuclear Operations, Inc.	)	
Indian Point Energy Center Unit 2	)	Docket No. 50-247
License Renewal Application	)	

PETITION FOR LEAVE TO INTERVENE, REQUEST FOR HEARING, AND CONTENTIONS

Friends United for Sustainable Energy, USA, Inc. (referred to hereinafter as FUSE, Stakeholders, Intervenors, or Petitioners), pursuant to 10 CFR § 2.309 (d) and (e), petition to intervene in the proceeding in response to the August 1, 2007 Notice of Opportunity for Hearing Regarding Renewal of Facility Operating License Number DPR-26 for an Additional 20-Year Period (72 FR 42134, August 1, 2007) concerning the Indian Point Energy Center License Renewal application of Entergy Nuclear Indian Point 2, LLC (referred to hereinafter as IP2 LLC) and Entergy Nuclear Operations, Inc. (referred to hereafter as Entergy Nuclear Operations) and (collectively referred to as the Applicant, or Licensee, or Entergy) to renew its operating license Nos. DPR-26 for Indian Point Energy Center Unit 2 ("IP2"), for

twenty years beyond the current expiration date of September 28, 2013. FUSE also requests a hearing under 10 C.F.R. §2.309(a).

I. PARTICIPATION AS A MATTER OF RIGHT

A. FUSE has standing

The standing requirement for Nuclear Regulatory Commission (NRC) adjudicatory proceedings derives from the Atomic Energy Act (AEA), which interest may be affected by the proceeding. 42 U.S.C. 2239(a)(1)(A). FUSE has standing on its own behalf and on behalf of its members. FUSE is a not-for-profit, New York State Corporation. FUSE is a nonpartisan sustainable energy policy think tank, whose purpose is to protect public health and safety. FUSE has members who live within the State of New York, New Jersey and Connecticut and who make their residences, places of occupation and recreation within fifty (50) miles of Indian Point. FUSE's address of incorporation is 34 Scenic Drive, Suffern, NY 10901, which is within nine miles of Indian Point and situated within the Plume Exposure Pathway (EPZ), also referred to as the "Peak Fatality Zone." The central office of FUSE is located at 21 Perlman Drive, Spring Valley NY, 10977, which is located within 11 miles of Indian Point and within the Indian Point "Ingestions Pathway" EPZ, , also referred to as the "Peak Injury" Zone.

FUSE also has numerous members that reside in the Indian Point immediate vicinity and throughout New York, New Jersey and Connecticut, whose concrete and particularized interests will be directly affected by this proceeding.

B. FUSE has standing on its own behalf

As stated in Ms. Susan H. Shapiro, Esq.'s attached declaration, Exhibit A, FUSE's headquarters are 21 Perlman Drive, Spring Valley, New York. Fuse's offices are within 11 miles of the Indian Point Entergy Center Unit 2 and within the "Ingestion Pathway EPZ," known as the "Peak Fatality Zone." FUSE's offices house the organization's records and material archives dating back six years. They also house an extensive technical book collection and FUSE furnishings and equipment. FUSE's offices also provide an operation center for the organization.

FUSE is reasonably concerned that the proposed Indian Point 2, LLC license could increase both the risk and the harmful consequences of an offsite radiological release. Furthermore, FUSE is concerned that the radiological contamination resulting from such a release would impact the value of its property, and interfere with the organizations rightful ability to conduct operations in an uninterrupted and undisturbed manner. *Id.* Certainly, any evacuation would severely disrupt and damage FUSE operations. *Id.* FUSE

therefore qualifies for intervention pursuant to 10 C.F.R. § 2.309(d).

FUSE also qualifies for discretionary intervention. 10 CFR § 2.309(e). FUSE's participation may reasonably be expected to assist in developing a sound record. It is well versed in the field of nuclear energy and safety. FUSE's constituency represents members who have participated in numerous Nuclear Regulatory Commission proceedings and public meetings. The nature of FUSE's interests is not only its members' (and its own) property interests but the public interest. In particular FUSE is a lead member of the Indian Point Safe Energy Coalition (IPSEC), a broad coalition of 70 other free standing organizations. The Stakeholders representing this filing also represent the 20 million resident Stakeholders within 50 miles of Indian Point.

FUSE can provide local insight that cannot be provided by the Applicant or other procedural parties. FUSE's members are Indian Point 2 neighbors. In addition, as established in this proceeding, this proceeding may have significant affect on FUSE and its members. Its members are Indian Point 2's neighbors. FUSE therefore qualifies for discretionary intervention. 10 C.F.R. § 2.309(e).

FUSE is entitled to a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G, because FUSE has standing, and in the Petition herein to Intervene and Formal

Request for Hearing, FUSE raises substantial issues of fact and law that meet the requirements of 10 CFR §2.310 (d).[1]

C. FUSE has Representational Standing

Declarations of Mr. Sherwood Martinelli contained in Exhibit B; Ms. Julie Gottesman contained in Exhibit C; and Mr. Gary Shaw, contained in Exhibit D demonstrate that FUSE members reside within the immediate vicinity of Indian Point. FUSE's members live less than fifty miles, and many less than ten miles from Indian Point 2, and are within its Emergency Planning Zone, and subject to radiological contamination, evacuation, loss of property, or other harms in the event of any mishap at the plant. *Id.* Members also use and enjoy the segment of the Hudson River adjacent to the Indian Point 2 on professional and personal bases. Declarations of Mr. Andrew Y. Stewart, Exhibit E, Mr. Timothy Englert, Exhibit F, and Ms. Jeanne Shaw, Exhibit G. The Hudson River is the receiving water body for any continued thermal discharge. *Id.* Declaration of Mr. Robert Jones, Exhibit H.

FUSE, an organizational and professional Intervenor, believes that its members' interests will not be adequately represented without this action to intervene, and without the opportunity to participate as full parties in this

[1] Although FUSE meets the requirements of 10 CFR §2.310(d) for a full adjudicatory hearing on all contentions it raises, FUSE does not concede the procedures of 10 CFR §2.310 which restrict use of full adjudicatory hearing procedures are lawful and reserves the right to challenge, in an appropriate legal forum, these procedures, as applied to FUSE in this case, should that be necessary to permit FUSE to fully adjudicate the important nuclear safety and environmental issues it raises.

proceeding. If the new superseding license for Indian Point (IP2) is granted without first resolving the Petitioner's safety concerns, this nuclear power installation may operate unsafely and pose an unacceptable risk to the environment and to the health, safety, and welfare of FUSE's members and the Stakeholders who live, recreate, and conduct business within its vicinity.

An organization has standing to sue on behalf of its members when a member would have standing to sue in his or her own right, the interests at issue are germane to the organization's purpose, and participation of the individual is not necessary to the claim or requested relief. *Hunt v. Washington State Apple Advertising Commission*, 432 U.S. 333, 343 (1977). As the Commission has applied this standard, an individual demonstrates an interest in a reactor licensing proceeding sufficient to establish standing by showing that his or her residence is within the geographical-area that might be affected by an accidental release of fission products. This "proximity approach" presumes that the elements of standing are satisfied if an individual lives within the zone of possible harm from the source of potential fission product release.

As is demonstrated by the above discussion and attached declarations, the members represented by FUSE all have standing in their own right. The issues of public health and safety are germane to FUSE's purposes. Also, the

individual participation of the members is not necessary to the claims or requested relief. Proximity [to a facility] has always been deemed to be enough to establish the requisite interest to confer standing. The Commission's "rule of thumb" in reactor licensing proceedings is that "persons who reside or frequent the area within a 50-mile radius of the facility" are presumed to have standing. *Sequoyah Fuels Corp.*, 40 NRC 64, 75 n.22 (1994); See also, *Duke Energy Corp.*, 48 NRC 381,385 n.1 (1998).

D. FUSE Meets Prudential Standing Requirements

In addition, Courts have created a prudential standing requirement that if a petitioner's interests fall within the "zone of interests" protected by the statute on which the claim is based. *Bennett v. Spear,* 520 U.S. 154, 162(1997). The Atomic Energy Act and NEPA, the statutes at issue here, protect the same interests of protecting public health and safety, that are held by FUSE's members, and furthered by FUSE's purpose.

II. FUSE DOES NOT WAIVE ITS RIGHTS TO SUBMIT SUPPLEMENTAL CONTENTIONS AND AMEND THE CONTENTIONS SET FORTH HEREIN, AND TO OTHER PROCEDURAL MATTERS

A. Right to supplement and amend contentions is not waived.

Regardless of the procedural violations of the Federal Administrative Procedures Act by the Applicant in submitting the License Renewal Application (LRA) and by the Nuclear Regulatory Commission in not

rejecting the LRA, FUSE is submitting a statement of the contentions that reflect the concerns of the Stakeholder community and should be accepted for hearing by the Nuclear Regulatory Commission on behalf of FUSE's members and broad constituency. The contentions submitted herein should not be deeded to waive FUSE's right to submit further contentions in the future or amend the contentions set forth herein. Further, FUSE reserves its right to submit additional contentions, and amend the contentions set forth herein.

B. Efficiency of Cross Examination of Expert or Fact Witnesses

The most efficient manner by which statutory rights can be exercised is to allow both depositions and live testimony to the extent the issues are not fully developed during discovery. Although not specifically mentioned in 10 CFR §2.102, cross-examination of witnesses will be more efficient when possible for FUSE and the Applicant to submit cross-examination outlines five days before the hearing, to alert each witness to the subjects which the parties will explore.

FUSE has the right to seek production of documents, if for no other reason than production of documents will facilitate interrogation of witnesses and narrow the scope of their examination. Otherwise, witnesses will be asked questions about issues which are addressed in documents which either are not

present during the interrogation or the analysis of which will require a hiatus in the interrogation.

Relevant documents and cross-examination outlines are hereby requested to be submitted by all parties wherever possible, at least five days in advance such that the witness may be prepared to fully answer the questions posed.

C. FUSE contends that the Nuclear Regulatory Commission and Applicant have had and will continue to have ex parte communications in violation of the requirements of Title 5, Part 1 Chapter 5 subchapter 11 § 557. Ex parte communication by the parties shall adhere in the strictest sense to the requirements of Title 5, Part I Chapter 5 subchapter II, §557.

The Stakeholders request that the NRC follows the regulations with regard to ex parte communications with the Applicant as required by Title 5, Part 1, Chapter 5 subchapter II§557. The sections that have particular relevance are provided below. In any agency proceeding which is subject to subsection (a) of this section, except to the extent required for the disposition of ex parte matters as authorized by law:

(i) No interested person outside the agency shall make or knowingly cause to be made to any member of the body comprising the agency, administrative law judge, or other employee who is or may reasonably be

expected to be involved in the decisional process of the proceeding, an ex parte communication relevant to the merits of the proceeding;

(ii) No member of the body comprising the agency, administrative law judge, or other employee who is or may reasonably be expected to be involved in the decisional process of the proceeding, shall make or knowingly cause to be made to any interested person outside the agency an ex parte communication relevant to the merits of the proceeding;

(iii) A member of the body comprising the agency, administrative law judge, or other employee who is or may reasonably be expected to be involved in the decisional process of such proceeding who receives, or who makes or knowingly causes to be made, a communication prohibited by this subsection shall place on the public record of the proceeding:

(A) All such written communications;

(B) Memoranda stating the substance of all such oral communications; and

(C) All written responses, and memoranda stating the substance of all oral responses, to the materials described in clauses (i) and (ii) of this subparagraph

(iv) Upon receipt of a communication knowingly made or knowingly caused to be made by a party in violation of this subsection, the agency, administrative law judge, or other employee presiding at the hearing may, to the extent consistent with the interests of justice and the policy of the

underlying statutes, require the party to show cause why his/her claim or interest in the proceeding should not be dismissed, denied, disregarded, or otherwise adversely affected on account of such violation; and

(v) The prohibitions of this subsection shall apply beginning at such time as the agency may designate, but in no case shall they begin to apply later than the time at which a proceeding is noticed for hearing unless the person responsible for the communication has knowledge that it will be noticed, in which case the prohibitions shall apply beginning at the time of his acquisition of such knowledge.

(vi) Therefore the Nuclear Regulatory Commission must abide by these regulations throughout the License Renewal Application proceedings and cease having ex parte communications with the Applicant, with regard to the License Renewal Application.

III. FUSE SUBMITS TWENTY-SIX ADMISSIBLE CONTENTIONS

A. Applicable Legal Standards to Specific Contentions

Proposed contentions must satisfy six requirements of 10 C.F.R. § 2.309(f)(1). This rule is intended to ensure that the "full adjudicatory hearings are triggered only by those able to proffer at least some minimal factual and legal foundation in support of their contentions." *Duke Energy Corporation (Oconee Nuclear Station, Units 1, 2 and 3)*, 49 N.R.C. 328, 334 (1999)

emphasis added. Sections (1) through (6) below summarize the requirements of § 2.309(f)(1).

1. Specifically State the Issue of Law or Fact to be Raised

Section 2.309(f)(i) requires a specific statement of issue of law or fact to be raised or controverted.

2. Briefly explain the Basis for the Contention

Section 2.309(f)(ii) requires a brief explanation of the contention.

3. Contentions must be within the scope of the Proceeding

Section 2.309(f)(iii) requires a petitioner to demonstrate that the issue raised in the contention is within the scope of the proceeding.

4. Contentions Must Raise a Material Issue

Section 2.309(f)(iv) requires "that the issue raised in the contention is material to the findings the Nuclear Regulatory Commission must make to support the action that is involved in the proceeding." Section 2.309(f)(iii) requires the petitioner to "demonstrate that the issue raised in the contention is within the scope of the proceeding."

(i) Scope of Environmental Review

The scope of the Nuclear Regulatory Commission's environmental review in the context of a license renewal proceeding is defined by 10 CFR Part 51 and by NRC's "Generic Environmental Impact Statement for License

Renewal of Nuclear Plants" (NUREG-1437 (May 1996). Some environmental issues are resolved generically for all plants, and such issues – classified in 10 C.F.R. Part 51, Subpart A, Appendix B as "Category 1" issues – are normally beyond the scope of a license renewal hearing. In the Matter of *Florida Power & Light Company (Turkey Point Nuclear Generating Plant, Units 3 and 4),* 54 NRC 3,15; 10 CFR § 51.53(c)(3)(i). The remaining issues in Appendix B, which are designated as "Category 2" issues, are issues for which (1) the applicant must make a plant-specific analysis of environmental impacts in its Environmental Report, 10 CRF § 51.53(c)(3)(ii), and (2) the NRC Staff must prepare a supplemental Environmental Impact Statement, 10 CFR § 51.95(c). Contentions concerning Category 2 issues are within the scope of license renewal proceedings. *Turkey Point Nuclear Generating Plant, Units 3 and 4,* 54 NRC at 11-13.

(ii) Scope of Safety/ Aging Management Review

10 CFR 54.4 sets forth the scope of review concerning safety issues in a license renewal proceeding. The safety review "is confined to matters relevant to the extended period of operations requested by the applicant," and focuses on the plant systems, structures, and components "that will require an aging management review for the period of extended operation," or "are subject to an evaluation of time-limited aging analyses." *Duke Energy Corp. (McGuire*

Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1, 2 and 3), 56 NRC 358, 363-64 (2002).

The NRC has emphasized that the level of inspection and testing related to age-management over the extended license term is one of the core issues addressed by the license renewal proceeding:

> Part 54 centers the license renewal reviews on the most significant overall safety concern posed by extended reactor operation – the detrimental effects of aging. By its very nature, the aging of materials 'becomes important principally during the period of extended operation beyond the initial 40-year license term,' ... Adverse aging effects can result from metal fatigue, erosion, corrosion . . . and shrinkage. Such age-related degradation can affect a number of reactor and auxiliary systems . . . Indeed, a host of individual components and structures are at issue. See 10 CFR 54.21(a)(1)(i). Left unmitigated, the effects of aging can overstress equipment, unacceptably reduce safety margins, and lead to the loss of required plant functions, including the capability to otherwise prevent or mitigate the consequences of accidents with a potential for offsite exposures.

Accordingly, Part 54 requires renewal applicants to demonstrate how their programs will be effective in managing the effects of aging during the proposed period of extended operation. Applicants must identify any additional actions, i.e. maintenance, replacement of parts, etc., that will need to be taken to manage adequately the detrimental effects of aging. Adverse aging affects are generally gradual and thus can be detected by programs that

ensure sufficient inspections and testing. *Turkey Point Nuclear Generating Plant, Units 3 and 4,* 54 N.R.C. 3, 7-8 (2001)(internal citations omitted).

5. Contentions Must be Supported by Facts or Expert Opinions

Section 2.309(f)(v) requires "a concise statement of the alleged facts or expert opinion which support the Petitioner's position on the issue and on which the petitioner intends to rely at hearing, together with references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue." An Intervener is not required to prove its case at the contention filing stage: "the factual support necessary to show that a genuine dispute exists need not be in affidavit or formal evidentiary form and need not be of the quality as that is necessary to withstand a summary disposition motion." Statement of Policy on Conduct of Adjudicatory Proceedings, 48 N.R.C. 18, 22 n.1 (1998), *citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process,* Final Rule,10CFR54, F.R. 33168, 33171 (Aug. 11, 1989). Rather, petitioner must make "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate." *In Gulf States Utilities Co.,* 40 NRC43, 51 (1994), *citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process,* Final Rule, 10 CFR,54 F.R. 33168, 33171 (Aug. 11, 1989).

6. Contentions Must Raise A Genuine Dispute Of Material Law Or Fact

Section 2.309(f)(vi) requires that petitioner:

> Provide sufficient information to show that a genuine dispute exists with the applicant/licensee on a material issue of law or fact. This information must include references to specific portions of the application (including the applicant's environmental report and safety report) that the petitioner disputes and the supporting reasons for each dispute, or, if the petitioner believes that the application fails to contain information on a relevant matter as required by law, the identification of each failure and the supporting reasons for the petitioners belief.

All that is needed is "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate." *In Gulf States Utilities Co.,* 40 NRC 43, 51 (1994), *citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process,* Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989).

B. FUSE asserts that the Applicant and the federal regulator made procedural violations of the Administrative Procedures Act, Subchapter II—Administrative Procedures. This resulted in substantive violations of the license renewal application submitted by the Applicant and accepted by the federal regulator.

The Applicant violated federal rule 10 CFR §54.4 when it submitted a single incomplete, inadequate and incorrect License Renewal Application that was in violation of specific regulatory requirements 10 CFR §54.4, which substantially affected three distinctly different nuclear facilities. Under 10CFR§54.17(d) "filing of application" an Applicant for a renewed license may submit an application for other kinds of licenses."

Therefore Indian Point 2 LLC and Entergy Nuclear Operations cannot file one application for the same license, nor can one application for three separate companies be filed for two separate licenses.

This rule does not mean, however, that multiple Applicants or licensees can file for a single license.

Each facility affected by this LRA is docketed individually, has distinct DPR numbers, was constructed by different Architect/Engineers under different General Design Criteria, and has different owners for most of each facility's operating history.

Responsive to the Administrative Procedures Act, the charter of the Atomic Energy Licensing Board as published in the Federal Register 37 FR 28,710 (1972) and the Commissioners regulations contained in 10CFR 2.104, 2.300, 2.303, 3.311, 2.318, and 2.132 may be interpreted to include contested issues in NRC licensing adjudications falling into two generic categories: (1) safety/technical issues arising under the Atomic Energy Act; and (2) environmental issues arising under the National Environmental Policy Act (NEPA). To renew the facility's operating license for an additional 20 years beyond its original 40-year license, the underlying application ***must include detailed analyses of the potential safety issues and environmental impacts posed by operating the plant for an additional 20 years.*** Members of the

public, state and local governments, and citizen organizations opposing the application can petition to intervene to contest the adequacy of the application's safety and/or environmental analyses.

The process for license renewal is sufficiently procedurally complex, and technically detailed to support regulatory rules for one LRA for each facility. The NRC technical staff (an agency entity entirely separate from the Atomic Safety License Board) conducts a thorough review and analysis of the technical and safety aspects of the application, and subsequently issues a Safety Evaluation Report that describes the staff's review and related findings. The staff also conducts a similar review on the environmental side, which typically results in the preparation of a full Environmental Impact Statement (EIS). Because major licensing actions generally require an EIS, Licensing Board cases regarding such activities usually have a significant National Environmental Protection Act component in addition to safety issues. In addition, Intervener Petitions and Requests for Hearings must be reviewed and adjudicated. Therefore by the Applicant co-mingling LRA's for two unique plants, LLCs and licenses, the Applicant further complicates the proceedings, thereby reducing the NRC's ability to conduct comprehensive, focused oversight for each individual facility.

Indian Point Unit 1 (Unit 1) is not even cited under the application, however it is substantially affected and affects the operations of Unit 2, in spite of it having been shut down for 33 years. This violation creates an avalanche of mixing of safety, technical and environmental issues caused by co-mingling, which introduce substantial additional complexity in the renewal proceedings. By failing to include Indian Point 1 components and systems in the LRA, the Applicant defeats the Stakeholders rights of Intervention and Hearings, promulgated under the Federal Administrative Procedures Act, with regard to the Indian Point 1 components and systems.

This egregious action alone by Entergy forestalls the publics' rights promulgated under the federal Administrative Procedures Act to adjudicate the proper decommissioning and remediation of the Unit1 site.

Therefore the NRC must deny the Applicant's LRA as being incomplete, inaccurate, incorrect and inadequately submitted.

IV. CONTENTIONS

A. Contentions 1 through 5: The Applicant violated the Administrative Procedures Act in bypassing the Code of Federal Regulations (CFR) and instead used trade guidance for Indian Point 2 instead of General Design Criteria for current design, and the current operating license with regard to the Applicant's LRA for an additional 20 years of operation.

Issue Statement: The regulatory rules for obtaining a new superseding license, as delineated in the code of federal regulations, specifically rules under 10 CFR 54, "License Renewal" and in particular, aging management as delineated under 10CFR54.21, were set aside by Applicant in lieu of suggested criteria promulgated by the trade industry. The Applicant misrepresented the specific General Design Criteria which formed the basis of the Safety Evaluation Report granting the Unit 2 operating license, and subsequently remained in violation of the terms of its operating license and with federal rules for four decades, and never corrected the obvious error—placing economics ahead of the health and safety of the public.

The Applicant, as well as the federal agency, willfully and knowingly violated the Administrative Procedures Act, and as a result now has prostituted the license renewal application submittal, contents, acceptance and approval for Indian Point Unit 2. The Aging Management Programs proposed by the Applicant are based upon misrepresentations of the actual general design criteria to which Indian Point 2 was licensed. The as-built construction of the facility does not comply with the safety evaluation report , the operating license or to the code of federal regulations.

The U. S. Nuclear Regulatory Commission (NRC) is currently assessing the need to review the 41 older nuclear power plant units referred to

as the Systematic Evaluation Program Phase III (SEP-III) plants. Generic Safety Issue (GSI) 156-6.1 (R. Emrit, et al., 1993) deals with whether the effects of pipe break inside containment have been adequately addressed in these plants' designs. The NRC originally evaluated a majority of the SEP-III plants before they issued Regulatory Guide (RG) 1.46 in May 1973 (AEC, 1973b). Although the NRC reviewed these plants, there is a potential lack of uniformity in those reviews due to the absence of documented acceptance criteria. The NRC is now attempting to assess the impact of not having such criteria in place.

The extent of the violations are breathtaking, and involve a substantial prima facie breach of Administrative Procedures Act (APA) by the Federal Agencies over almost four decades for Indian Point 2. Beginning in 1968, the Nuclear Regulatory Commission acted in direct defiance of the Administrative Procedures Act by approving Amendment Nine of the Operating License, (contained in exhibit I) in which the Licensee acknowledged commitments to *trade comments* to draft General Design Criteria for its new plant. In addition, the Licensee committed to trade comments to the proposed General Design Criteria, and erroneously claimed that the trade organization comments were published in the Federal Register

for public comment in July, 1967, when in fact they were never properly published (see Exhibit J)

The Licensee claimed adherence to a General Design Criteria required for the licensing of Indian Point 2 facility, and committed to such General Design Criteria in the 1970 SER. In actuality, the plant design, programs and procedures *were licensed to trade industry-endorsed commentary* as opposed to the General Design Criteria for the LRA and subsequently approved by the Atomic Energy Commission under the 1970 Safety Evaluation Report (See Exhibit K) bypassed the federal rules as found under the rule making process. The draft GDCs were published and approved for use more than 13 months prior. This fundamental failure of oversight by the regulator was subsequently set aside and festered, while the commission quietly authorized by retroactive fiat that the licensing process proscribed under federal rules for Indian Point 2 could remain in violation of law. This series of events is evidenced by close examination of documents cited or submitted in the applicant's LRA. The commission dealt with the design basis and license failures with a stroke of a pen in 1992. (see exhibit L)

The table below best provides the chronology as well as the facts, and the implications to the renewal license application fidelity. In simplest terms the Licensee and NRC with the acceptance of the GDC defined in

Amendment 9 to the original application for license accepted a draft industry GDC in place of the actual GDC for IP2.

Date:	Docketed Activity	Reference	Implications to fidelity of the License Amendment
November 22, 1965	Early draft General Design Criteria published by AEC for comment	November 22, 1965 Press release from AEC. No FR notice	For consideration by Con Ed in decision to Construct Indian Point 2
October 14,1966	By application dated December 6, 1965, and amendments thereto (the application), the applicant applied for the necessary licenses to construct and operate a nuclear power reactor at the applicant's site at Indian Point, Village of Buchanan, Westchester County, New York.	The Commission, after a public hearing and after an initial decision by the Atomic Safety and Licensing Board (the Board), established by the Commission, issued Construction Permit CPPR-21 for this facility	The application was evaluated by the Commission's regulatory staff and independent Advisory Committee on Reactor Safeguards (ACRS), both of which concluded that there is reasonable assurance that the facility could be operated at the proposed site without undue risk to the health and safety of the public. On October 14,1966,
July 11, 1967	AEC publishes draft General Design Criteria under federal rule making processes.	Federal Register 32 FR 10213	Note that the draft GDCs were never made a part of Appendix A of 10CFR50.
October 2, 1967	Atomic Industry Forum, a trade organization provides significant comments regarding draft GDCs published.	Provided directly to Atomic Energy Commission without publication in the federal register	AIF general proposed removal of conservatism in draft General Design Criteria. **These changes were never approved by the AEC.**
October 15, 1968	Former owner of Unit 2 submits Amendment 9 of application of license	AEC Docket No. 50-247-- correspondence from Con Ed to Director of Division of Reactor Licensing Atomic Energy Commission	Facility that was now more than 2 years into construction was being constructed following unapproved trade documents – however, the letter states on page 1.3-1 that the unapproved "general design criteria tabulated explicitly in this report comprised of the proposed AIF versions of the criteria issued for

Date:	Docketed Activity	Reference	Implications to fidelity of the License Amendment
			comment in July 1967."
February 1970		See January 28, 1971 NRC discussion of AIF GDC comments.	The staff met with an ad hoc AIF group, which included representatives of reactor manufacturers, utilities and architect engineers to discuss the revised General Design Criteria. The comments of this group were reflected in a June 4, 1970 draft of the revised General Design Criteria that was forwarded to the AIF for comment. The AIF forwarded comments and stated it believed the criteria should be published as an effective rule after reflecting its comments. These comments have been reflected in theGeneral Design Criteria in Appendix "A".
November 16,1970	Safety Evaluation Report Commission grants operating license based upon amendments 9-25 of application for license by Con Edison.	Incorporated License amendments 9-25 to the application and the FFDSAR -includes ALSB, ACRS review et al.	"Our technical safety review of the design of this plant has been based on Amendment No. 9 to the application, the Final Facility Description and Safety Analysis Report (FFDSAR), and Amendments Nos. 10-25, inclusive. All of these documents are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D.C. The technical evaluation of the design of this plant was accomplished by the Division of Reactor Licensing with assistance" from the Division of Reactor Standards and various consultants to the AEC. This document gave them authority to operate the facility under the draft GDCs but **without the AIF comments specifically for the** Reactor Protection and Control System.

Date:	Docketed Activity	Reference	Implications to fidelity of the License Amendment
			As noted, "Specifically, for the reactor protection system instrumentation for -Indian Point Unit 2 is the same as that installed- at the Ginna plant. The adequacy of the protection system instrumentation was evaluated by comparison with the Commission's proposed general design criteria published on: July 11, 1967, and the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 2 is acceptable".
February 20 1971 through July 11 1971	Formerly Draft GDCs are approved Final GDCs and become part of Appendix A to 10CFR50. They are amended the same year.	Published in FR. on February 20 1971, and amended on July 11, 1971	These are the first legal standards for which the plant is required to comply or under federal rules, or be granted an exemption.
November 4, 1971	A third modified construction permit was issued for Units #1 and #2. The proposed relocation of the intake structures by Con Edison was a significant improvement and entered into this decision.		The USAEC is urged to require Consolidated Edison to establish a firm schedule for implementing this proposed modification because of changes in the design of the adjustable discharge ports and slide gates.
September 28, 1973	Unit 2 Operating License Received		SER states that the plant is licensed to 1967 draft general design criteria **without endorsement of AIF comments.**
Commission issues a confirmatory order on February 11, 1980	Unit 2 FSAR dated June 2001 states that the detailed results of the order indicate that the plant is in compliance with the then current General Design Criteria established in 10CFR50		The commission concurred on January 1982.

Date:	Docketed Activity	Reference	Implications to fidelity of the License Amendment
	Appendix A.		
September 18, 1992	SECY 92-223, "resolutions of deviations identified during the systematic evaluation program"	Letter to James Taylor, Executive Director for Operations	The Commission approved the staff proposal in which the plant will not be required to comply with federally approved General Design Criteria, if construction permits were issued prior to May 2, 1971. This appears to be a clear and flagrant violation of the Administrative Procedures Act.
June 2001	Unit 2 FSAR states incorrectly that the General Design Criteria tabulated explicitly in the pertinent systems comprised the proposed trade organization general design criteria.	Section 1.3 General Design Criteria, Unit 2 UFSAR, and indicates under a footnote that the safety analysis report added trade organization comments in the change to the FSAR. (see footnote within Section 1.3.)	The license with collateral endorsement of the federal regulatory agency bypassed the administrative rules act, and thus reduced its commitments made to obtain its operating license to less than the minimum legal requirements of 10CFR50 Appendix A which were made law more than two years prior to the NRC granting the applicant an operating license for Unit 2. The reductions of margin and reasonable assurance of protection of the health and safety of the public were compromised for three decades, without the public understanding of the loss of margin in safety. Subsequently, the applicant (now Entergy) allowed the error to remain and is actually currently committing Unit 2 to trade organization design criteria.

The Licensee's failure to adhere to a legally enforceable General Design Criteria substantially reduces safety margins for safe plant operation, by severely reducing detection of and the consequential mitigation of

accident conditions resulting in substantial reduction in protecting the health and safety of the public.

The Nuclear Regulatory Commission continued this pattern of bypassing the Administrative Procedures Act in 1992, (see exhibit I), in which the regulator relieved the Applicant of *all c*ompliance enforcement to any General Design Criteria, without any attempt to abide by the Administrative Procedures Act.

The Commission belief that it could use guidance documents from trade organizations in lieu of rules as was adjudicated in *Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit No. 1) ("TMI")* ALAB-698, 16 NRC 1290, 1298-99 (October 22, 1982), affirming *LBP-81-59, 14 NRC 1211, 1460 (1981),* where it was established that the criteria described in NUREG-0654 were intended to serve solely as regulatory guidance, not regulatory requirements*)*. Indeed, the Commission's mere reference to NUREG-0654 in a footnote to 10 C.F.R. § 50.47 was found to be insufficient to incorporate that guidance document by reference as a part of a federal regulation, even if the Commission had intended to do so.

The Nuclear Regulatory Commission continues this approach today without any hint of complying with the rules of the Administrative Procedures Act (APA). In summary, the Applicant is obligated to meet the requirements

of the General Design Criteria as published on July 11, 1967. In fact, the Applicant falsely states that it is in compliance on page 3 of the LRA. Indian Point 2 LLC plant was designed, constructed and is being operated on the basis of the proposed General Design Criteria, published July 11, 1967. Construction of the plant was already underway when the Final Facility Description and Safety Analysis Report was filed on December 4, 1970, and when the Commission published its revised General Design Criteria in February 1971, and final version of the General Design Criteria in July 1971, which included the false statement, "As a result, we did not require the applicant to reanalyze the plant on the basis of the revised criteria. However, our technical review assessed the plant against the General Design Criteria now in effect and we have concluded that the plant design conforms to the intent of these newer criteria."

The Applicant was not in compliance with 10CFR50 Appendix A then, and is not in compliance with 10CFR50 Appendix A now, as provided in current 2006 Unit 2 UFSAR submitted as a part of its relicensing application.

Subsequent to the issuance of the Operating License, the Nuclear Regulatory Commission issued many Bulletins, Orders, Generic Letters, and Regulatory Guides. Most of the Regulatory Guides address the Nuclear Regulatory Commission's interpretation of the meaning of the requirements of

the 1971 General Design Criteria. Inference could be made that regardless of the legal basis of these orders, if one accepts them as legal, one must also accept the legal requirement of compliance to the specific relevant 1971 General Design Criteria. However, the process clearly violated the Administrative Procedures Act regarding the incorporation by reference on regulations such as violation of 10CFR50.21[2], regarding equipment aging program scope by using a methodology that is entirely addressed under NUREGS prepared and promulgated outside rulemaking procedures and industry trade guidelines such as NEI 95-10 Rev. 6, each of which has no legal force. Neither public involvement nor the most fundamental steps required under the Administrative Procedures Act were adhered to by either the Applicant or the Federal Agency.

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[2] (a) Plant systems, structures, and components within the scope of this part are--

(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions--

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

(2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.

(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

(b) The intended functions that these systems, structures, and components must be shown to fulfill in § 54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this section.

[60 FR 22491, May 8, 1995, as amended at 61 FR 65175, Dec. 11, 1996; 64 FR 72002, Dec. 23, 1999]

Pursuant to section 3(a)(1) of the Administrative Procedure Act, 5 U.S.C. § 552(a)(1), as implemented by the regulations of the Office of the Federal Register, 10 CFR Part 51, no material may be incorporated into a rule by reference unless the agency expressly intends such a result, 10 CFR. § 51.9, requests and receives the approval of the Director of the Office of Federal Register, 10 CFR §§ 51.1, 51.3, and the Federal Register notice indicates such specific approval, 10 CFR § 51.9.

A brief review of statutory/regulatory construction confirms the method for incorporating Regulatory Guides . Here 10 CFR Part 50, Appendix E, n.1; NRC Staff Regulatory Guide 1.101, Rev. 2 (October, 1981) specifically endorses the incorporation by reference to the criteria and recommendations in NUREG-0654 as "generally acceptable methods for complying" with the standards in 10 CFR § 50.47. The NRC's emergency planning rules, however, include neither such a designation nor any express intention that NUREG-0654 be incorporated by reference.

In the absence of other evidence, adherence to NUREG-0654 may be sufficient to demonstrate compliance with the regulatory requirements of 10 CFR § 50.47(b). However, such adherence to NUREG-0654 is not required, because regulatory guides are not intended to serve as substitutes for regulations. *TMI, ALAB-698, supra, 16 NRC at 1298-99.* "Methods and

solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission." *Id.* at 1299, quoting *Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-644, 13 NRC 903, 937 (1981).* We believe the atomic licensing board erred in this decision. This error was confirmed in the recent ruling regarding storage of spent fuel requiring a NEPA proceeding compliance prior to the NRC approval. See *San Luis Obispo Mothers v. NRC 03-74628*

Examples include certain Regulatory Guides that provide requirements for post-accident monitoring of the TMI incident. These Regulatory Guides describe a method that the NRC staff considers acceptable for use in complying with the agency's regulations and delineate an acceptable means of meeting the General Design Criteria as contained in 10 CFR 50 Appendix A. More than 100 Regulatory Guides have been issued, amplifying the requirements of the General Design Criteria.

Consider the U.S. Nuclear Regulatory Commission (NRC) developed Regulatory Guide 1.97 to describe a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. Specifically, the method described in this Regulatory Guide relates to

General Design Criteria 13, 19, and 64, as set forth in Appendix A to Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities:

> Criterion 13, "Instrumentation and Control," requires operating reactor licensees to provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.
>
> Criterion 19, "Control Room," requires operating reactor licensees to provide a control room from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs). In addition, operating reactor licensees must provide equipment (including the necessary instrumentation), at appropriate locations outside the control room, with a design capability for prompt hot shutdown of the reactor.
>
> Criterion 64, "Monitoring Radioactivity Releases," requires operating reactor licensees to provide the means for monitoring the reactor containment atmosphere, spaces containing components to recirculate LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released as a result of postulated accidents. The licensee has responded to these communications and states compliance with these communications and makes a commitment in the UFSAR.

In these examples, the Applicant included the NUREG language in the FSAR, and by inference one could argue compliance in this case with General Design Criteria 1971. The Applicant could not, however, use the Aging Management Program to argue compliance with other cases, and certainly cannot use the program exclusively. The Applicant is potentially holding open

options that should be eliminated under the Aging Management Rule. (See Contention 3).

A dispositive example is "General Design Criteria" Criterion 35-Emergency core cooling:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is available) the system safety function can be accomplished, assuming a single failure.

See General Design Criteria 35, Final design criteria (10 CFR 50 appendix A approved 1971, (36 FR 3256, Feb 20, 1971)

<u>The IP2 Final Safety Analysis Report (FSAR) does not address Criterion 35 at all</u>. In neglecting to do so, the IP2 FSAR leaves the General Design Criteria meaningless in its intent to protect the health and safety of the public, and places the plant in clear violation of 10CFR50 Appendix A.

A detailed list of specific violations contained within 10 CFR Part 54 will be provided in supplemental submittal to this contention. An example is

(Contention 4) provided below from review of the limited material available to FUSE by the Licensee , and the regulator.

> Criterion 10, Reactor design, in which the reactor core and associated coolant, control, and protection systems must be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

FSAR Section 5.1.1.1.5, Reactor Containment substantiates the Criterion with the following additions:

> The containment structure shall be designed (a) to sustain, ***without undue risk to the health and safety of the public***, the initial effects of gross equipment failures, such as a ***large reactor coolant pipe break***, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to ***the extent necessary to avoid undue risk to the health and safety of the public. [italics added by Stakeholders]***

These additions provide latitude and judgment to the Applicant as to what the Architects and Engineers need to do in order to minimally satisfy the criteria ***but do not support the right for public review of the pertinent documents in a public forum.***

A brief review of Tech Spec requirements contained in Exhibit L confirms that the misrepresented statement in the FSAR regarding General Design Criteria for Unit 2 is followed through with improper implementation.

For example, Reactor Coolant Leakage. In LCO 3.4.13, reactor containment pressure leakage from primary to secondary systems ***is allowed in quantities up to 150 gallons per day***. Such quantities are much larger than reasonable limits implicit under General Design Criterion 35. This non-conservative quantity may have contributed to the root cause of the 2000 tube rupture accident and is intolerable as an acceptable quantity for age management of the RCS leakage.

A second example may be found in examination of General Design Criterion 45, through General Design Criterion 6.2.1.2. Inspection of Emergency Core Cooling System Criterion is the following: Design provisions shall, where practical, be made to facilitate inspection of physical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles. (General Design Criteria 45). ***Here the trade organization inserted the words "where practical"*** (see Exhibit N page 14).

The Applicant bypasses the rules, by failing to properly examine or replace reactor core internal components with known susceptibility to failure on multiple occasions. For example, the components such as baffle bolts that hold down springs, lower core barrel, and lower core plate are routinely UT or VT'd during outages and often replaced. (See exhibit P)The process involves a machine that typically removes and replaces bolts in an automated

procedure which adds two weeks to an outage. Despite the higher reliability of such a process, Indian Point 2 has chosen instead to rely on water chemistry tests which are meaningless for assessing bolt integrity. The reasoning behind the reliance on an inferior method of testing is financial: Water chemistry tests enable Indian Point 2 to substantially reduce lost revenue by shortening the outage time(some estimates are in the order of millions of dollars per outage day), despite the fact that the health and safety of the public is sacrificed. See exhibit P and declaration of Ulrich Witte, Exhibit Q. This is a prima facie violation of 10CFR50 Appendix A.

The Applicant attempts to placate the issue with the following words contained in the LRA, "To manage loss of fracture toughness, cracking, change in dimensions (void swelling), and loss of preload in vessel internal components, the site will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval." See section A.2.1.141 of the LRA report.

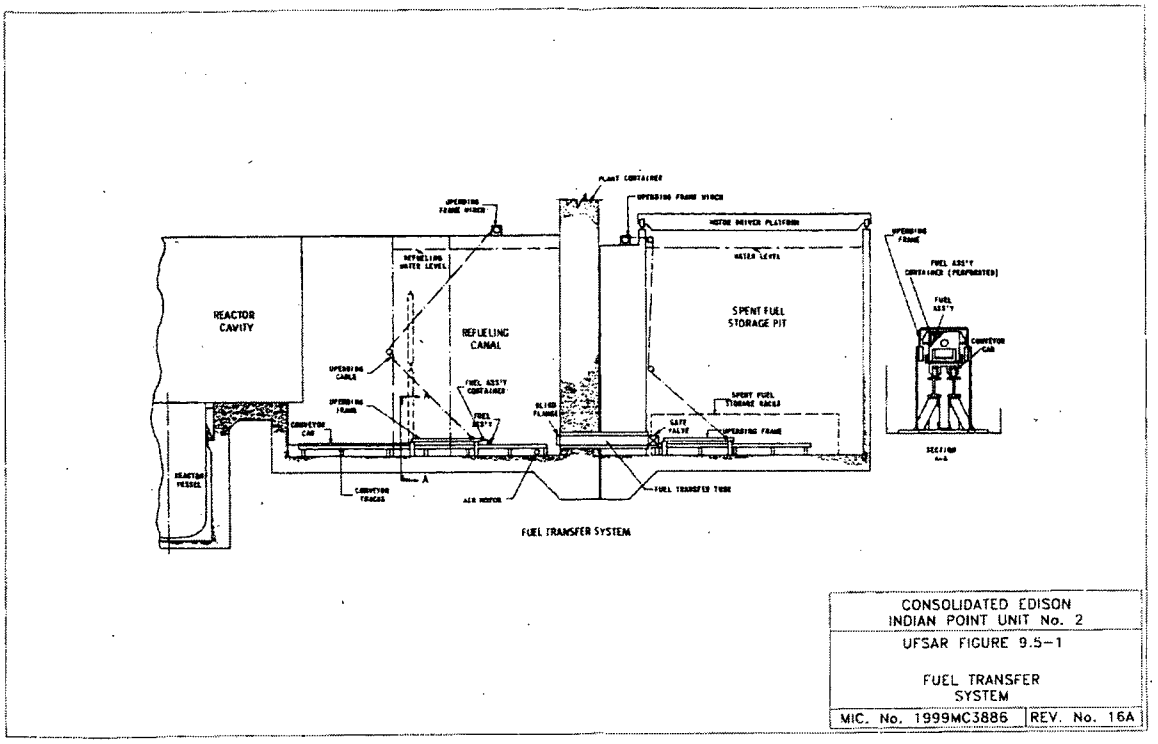
This language essentially removes this entire matter from the public's

right of input and participation. It is another example of "Agree to agree" and bypasses the procedures required by law through the Administrative Procedures Act.

Alternative methods that act as proposals to comply with the federal rules for license renewal represent guidance only, unless explicitly cited, and developed within the confines of the Administrative Procedures Act. The above examples meet the standards for specific contentions as cited above.

This serious and deliberate practice of rewriting federal code without public input is in clear violation of the Administrative Procedures Act and invalidates the plans proposed for the technical, safety, and environmental aspects of entire LRA, even setting aside the issues of a lack of completeness and vagueness of the description.

The misrepresentation has become routine, and the violations so acceptable, that the NRC only days ago published a notice regarding a leaking and aging 20-inch pipe, described as a "conduit" with a pinhole leak.



1. Misrepresentation does violence to the entire intent of the agency, and the Applicant's failure to comply with specific rules of 10 CFR 54, and further violates the Administrative Procedures Act. For example, the 20-inch "conduit" is not considered part of the Aging Management Program or part of the environmental program, and the lack of inspection and maintenance of it is not considered unlawful. See exhibit R, and we ask that this be considered Contention Number 5

The breadth and depth of these contentions are extreme. Even if each issue is classified in the narrow confines of the scope of the Rule (but not the GALL Report (see NUREG 1801 Rev. 1), the egregious conduct by the

applicant and the regulatory failure raises questions about any statement made in the LRA, or the Current Licensing Basis for Unit 2. The Current Design Basis for Indian Point 2 is unknown, unmonitored, and the material condition also unknown. These conditions associated with the CLB were the exact bases for permanent closure of Millstone Unit 1. These findings for Indian point 2 are clearly analogous, and a new superseding license should without question be denied.

For those issues raised here, no forum is available to adjudicate the magnitude of the misrepresentation and unlawful acts. FUSE questions how a Board selected by the Commission can be allowed to judge the acts of the very Commission that selected it (such as the 1992 letter contained in Exhibit M). The Administrative Procedures Act under chapter 5 provides for adjudication in the federal court for exactly this kind of broad unlawful act.

A. Contention 6: The License Renewal Application (LRA) fails to provide sufficient detailed information regarding technical, safety and environmental pendant issues as required by 10 CFR 2.309.

Issue statement: FUSE asserts that the Applicant's LRA has not met the threshold of providing explicit specific technical information as required under 10 CFR54, specifically with regard to Equipment Environmental and the Qualification Program, Flow–accelerated Corrosion Program.

The license renewal application submitted by applicant on May 3, 2007 and subsequently revised on June 22, 2007 fails to meet the threshold of providing explicit specific technical information as called for under 10 C.F.R. § 2.309, which plainly calls for *"how the applicant will comply with the requirements"* promulgated in CFR54.21 and requires both a complete description of each program and a description of how the applicant will specifically address Aging Management." In the LRA submitted by the Applicant , these threshold requirements, are not included, or provided other than with non-specific conclusory statements.

Specific examples of incomplete and inadequate technical information include, but are not limited to: the Equipment Environmental Qualification Program, the Flow-accelerated Corrosion Program, in which the Applicant provided a one paragraph description of its planned Aging Management Program, which essentially credited the current Flow-accelerated corrosion ("FAC) program with no further explanation. Here, the Applicant points to the present Current Licensing Basis ("CLB") as sufficient. This is an ambiguous and generic approach that is rejected under both NUREG 1801, and 10CFR54 as well. The rules require that a specific and particularized program define component and system scope, inspection criteria,

methodology, frequency and remediation commitments when acceptance criteria for FAC inspections are not met.

This contention is fundamentally material to the Indian Point License Renewal Proceedings as a matter of law. The Applicant's failure to comply with the 10 CFR54 rules setting forth Age Related Management Programs, makes it virtually impossible to review the legal or technical integrity regarding each of these programs. This raises fundamental and material issues to the entire LRA content as submitted by the Applicant.

B. Contention 7: Co-mingling three dockets, and three DPR licenses under a single application is in violation of C.F.R. Rules, Specifically 10 CFR 54.17 (d) as well as Federal Rules for Civil Procedure rule 11(b).

Stakeholders assert that the Applicant's single LRA for three distinct licenses and nuclear plants is a violation of 10 CFR 54.17(d), as well as the Federal Rules for Civil Procedure Rule 11(b), thereby causing the LRA review to be overly complex, unclear, and unduly confusing, and should be denied by the NRC.

The applicant has violated rule 10 CFR §54.17 (d), which states,

> *An applicant may combine an application for a renewed license with applications for other kinds of licenses.*

This does not mean or intend to mean that the Applicant can co-mingle two applications for two license renewals, for Indian Point 2 and Indian Point 3, into one LRA filing. To make things even more complicated, components of Indian Point 1, which has been shut down for 30 years, are used by Indian Point 2, therefore Indian Point 1's Safestor status must be incorporated by reference. IP2 and IP3 hold completely separate licenses to operate nuclear reactors. Each license is further held by a separately owned and controlled Limited Liability Corporation. In addition, the Applicant violates procedure governed by 10 CFR by not distinguishing the current Safestor status of Unit 1 decommissioning, and in fact seeking approval to make use of Unit 1 systems and/or components/infrastructure for extended operation of Unit 2, and to a lesser degree Unit 3.

Co-mingling applications is particularly material to Indian Point 2 and 3 given that each license has (1) separate dockets [50-247 and 50-286], (2) separate DPR numbers, (3) separate owners and License holders for most of their first 30 years of operation, and (4) separate Architect/Engineers.

The Nuclear Regulatory Commission itself at the annual assessment meeting has admitted the plants have entirely different histories, different design control and configuration management programs. The NRC held and

will continue to hold separate reviews to discuss each reactor licensees' separate issues prior to opening the meeting for public questions.

Indian Point 2 and Indian Point 3 had and continue to have distinctly different Current Licensing Bases (CLB), and have evolved away from each other via a multitude of different design modifications.

The Contention that the Applicant has wrongfully commingled and cojoined the applications for IP2 LLC and IP3 LLC is buffered and strengthened by the fact that the NRC itself has assigned separate onsite plant inspection teams to each individualized reactor.

Indian Point 2 has been repeatedly in "white status" for the past 10 years, and Indian Point Unit 3 was on the NRC's watch list during the 90s, while the plants have been subjected for over 30 years to different corrective action programs, and different design control programs; and each has its own set of active licensing commitments with respect to their Current Operating License and plant technical specifications.

C. Contention 8: The NRC violated its own regulations by accepting a single License Renewal Application made by the following parties: Entergy Nuclear Indian Point 2, LLC ("IP2 LLC") Entergy Nuclear Indian Point 3, LLC (" IP3 LLC"), and Entergy Nuclear Operations, LLC. (Entergy Nuclear Operations).

Issue Statement: FUSE asserts the ownership and legal liability associated with the superseding licensing is incomplete and inaccurate as described in

Entergy's application for renewal by not including holding companies that differ for each plant (see exhibit S, organizational figures), and that the NRC should not transfer the license from one LLC to another in the middle of a LRA review as announced by an entirely different holding company in Entergy's letter dated July 30, 2007, see exhibit S.

Based upon the documents submitted in the July 30, 2007 letter, the current license does not correctly describe the owners of the Unit 2 facility, the operators of the Unit 2 facility are not unambiguous and cause undue confusion of ownership regarding matters relevant to future decisions, especially concerning extended operations regarding the superseding license being proposed.

Even though named on the current operating license, Entergy Nuclear Operation Inc. cannot be a party to the LRA, and should not be named on the current operating license, because it lacks the necessary direct relationship between the Licensees and Entergy Nuclear Operations. Nor is Entergy Nuclear Operations, Inc. involved in daily operations or record keeping, in direct violation of 10CFR50.

Entergy Nuclear Operations is not currently the operator or direct owner of the license, and thus does not have direct control over the license, nor does it maintain records or additional records as required by 10CFR54.35 and 10CFR54.37.

In the case of Indian Point 2, the immediate owner is Entergy Nuclear IP2, LLC. This LLC is in turn owned by Entergy Nuclear Investment Company III, Inc., which is a wholly-owned subsidiary of Entergy Nuclear Holding Company #3 that, in turn, is a wholly-owned subsidiary of Entergy

Nuclear Holding Company. Entergy Nuclear Holding Company, Inc. is a direct subsidiary of Entergy Corporation. The NRC's own staff have expressed serious doubts as to the NRC's ability to hold a parent corporation responsible for the liabilities incurred by a subsidiary.

A particular concern is that each intervening LLC can act as a barrier to extending liability to the parent corporation that contains most of the assets. Several separate litigations, or a very large and complex single litigation would be required to pierce all the corporate veils back to the parent corporation with the bulk of the assets. (*Synapse Energy Economics, Inc Financial Insecurit*y pg 12 attachment see exhibit V).

Just two days prior to formal application acceptance by the Staff it was announced in the Federal Registry, Entergy Nuclear Operations filed for a transfer of Indian Point 2 license DPR-26 and Indian Point 3 license DPR-64 to Entergy Nuclear Operations, an indirectly related corporation, which would result in substantial reorganization of Entergy's corporate structure and LLC holdings, affecting the fiscal responsibility and liabilities of Indian Point 1, Indian Point 2 and Indian Point 3.

This whole overly complicated corporate structure overlay on top of another corporate structure overlay, is akin to Abbot and Costello's who's on first, and who's on second, but the humor dissolves when the questionable

motivation and the detrimental consequences to the health and safety of the public become apparent.

This overly complicated corporate structural overlay has severe consequences to reasonable assurances of health and safety of the public.

The motivation behind this requested license transfer is revealed when one reviews how the parent corporation of Entergy handled its fiscal liability with regards to Hurricane Katrina events by comparing the historical actions of the parent company and understanding how Entergy ducked the fiscal liability associated from the Katrina events.

In the aftermath of Katrina, Entergy New Orleans, a subsidiary of the The Entergy Corporation, filed for Chapter 11 bankruptcy, even though the parent corporation continued to have ample finances. This corporate hide and seek resulted in Entergy Corporation receiving massive government bailouts from taxpayers monies, while ratepayers in New Orleans experienced a substantial increase in energy costs.. (Exhibit W cite Rita King article).

The NRC has no statutory authority to require a licensee in bankruptcy to continue making safety-related or decommissioning expenditures or to pay retrospective Price-Anderson Act premiums. Therefore, any transfer of the licenses in the middle of an LRA proceeding brings into scope Entergy's entire corporate structure and complex financial qualification review to

continue operating the licenses during the license renewal period of 20 years. We claim this very issue as another non numbered contention.

Moreover, the timing of this transfer application creates the opportunity for the NRC staff to do less than an adequate review, as was found by the GAO in previous reviews performed {reference GAO report exhibit X}, and diverts the NRC staff's full attention from the technical requirements and assurances of public health and safety during the LRA reviews, to devote substantial resources and attention for a complex financial qualification review.

The General Accounting Office has found that the NRC has done an inadequate analysis regarding the fiscal responsibility during license transfers in the past, affecting commitments or lack thereof, including but not limited to such items as the decommissioning funds (specifically relevant to Unit 1 and Unit 2 license renewal). The proposed transfer of the license materially affects the fiscal resources and clear liability for each of the three Indian Point Units.

If the NRC reviews and approves this proposed license transfer in the middle of the LRA review, it will add undue confusion and complication resulting in harm to the Stakeholder's rights, in turn causing potential harm to the public's health and safety.

B. Contention 9: The Decommissioning Trust Fund is inadequate and Entergy's plan to mix funding across Unit 2, 1 and 3 violates commitments not acknowledged in the application and 10 CFR rule 54.3.

Issue Statement: FUSE asserts that the Applicant's decommissioning trust fund balances are inadequate and insufficient to properly decommission the site, as required by 10CFR 54.3 to restore the site. Therefore, due to the inadequacy of the decommissioning trust funds, the NRC cannot approve a new superseding license for an additional 20 years.

Indian Point 2 has insufficient decommissioning trust fund balances, as required by 10 CFR 50.75, to restore the Indian Point site, including removal of underground radioactive contamination in the bedrock under the plant.

Per NRC Section PART 50 Sec. 50.75: Reporting and recordkeeping for decommissioning plan Indian Point's decommissioning funds are inadequate to clean up the bedrock site from the ongoing underground leaks. The costs for complete decommissioning and cleanup of the site must be adjusted to reflect significant changes in the contamination streams including the large underground radioactive leaks. However the Applicant has not evaluated, calculated or considered the actual decommissioning funds required to decontaminate the site in light of ongoing massive underground radioactive effluent and leaks.

2. Basis for Contention

The Indian Point 2 decommissioning fund has not been adjusted to take into consideration the enormous, underground radioactive contamination accidentally discovered in 2005. The current decommissioning plan for aging management of the plant is inadequate to clean up the bedrock site and is not addressed in the Applicant's LRA. The costs for complete and correct decommissioning and cleanup of the site must be adjusted to reflect the large underground radioactive leaks, as required by:

> Section PART 50 Sec. 50.75 (2) (e)(1)(v); any modifications occurring to a licensee's current method of providing financial assurance since the last submitted report; and any material changes to trust agreements.... or where conditions have changed such as:
>
> (iii) The current situation with regard to disposal of high-level and low-level radioactive waste;
> (iv) Residual radioactivity criteria;
> (v) Other site-specific factors which could affect decommissioning planning and cost;
> (1) Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site.
> These records may be limited to instances when significant contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records must include any known information on identification of involved nuclides, quantities, forms, and concentrations., or certification is used.

It has been acknowledged by the NRC that numerous systems, structures and components can experience undetected radioactive leaks over a prolonged period of time and that "relatively large volumes of contamination above the decommissioning release limits" can result in "notable increases in remediation time and costs" in the sums of hundreds of millions of present value dollars. *NRC's _Liquid Radiation Release Lessons Learned Task Force Final Report,* ML062650312 2006-09-013.4.3 The past and present leaks at Indian Point 2 provide indicia of continued and future leaks. In 2006 Don Mayer, Director of Special Projects for Entergy said that "The underground area of the Indian Point site has contaminated water that is 50 to 60 feet deep, ...and there is also another area, or underground plume, that is about 30 feet wide by 350 feet long."

3. Contention is within scope in the licensee renewal process

In the Matter of Power Authority Of The State Of New York And Entergy Nuclear Fitzpatrick LLC, Entergy Nuclear Indian Point 3 LLC, And Entergy Nuclear Operations, Inc. (*James A. FitzPatrick Nuclear Power Plant and Indian Point Nuclear Generating Unit No. 3*) Docket Nos. 50-333-LT and 50-286-LT regarding the license transfer to Entergy, the Nuclear Regulatory Commission held that decommissioning shortfall "did not fall within the

scope of this license transfer proceeding, as Entergy Indian Point was not seeking in its application to renew or extend the Indian Point 3 operating license, nor does its pending application assume such a request.

The Commission further states, "that regarding decommissioning Stakeholders have the right to seek intervenor status in any application for license renewal or license extension that Entergy Indian Point may file." Therefore, based on the Commission's own decision, the issue of whether there are adequate decommissioning funds is within scope of the licensing renewal proceedings.

4. Contention raises a material issue of fact or law

The method of cost analysis of adequate decommissioning funds must be clearly stated in the LRA. The Applicant's LRA fails to outline an adequate decommissioning and clean up plan in light of the large amounts of underground radioactive waste, for which the source has not yet been identified, and therefore the extent of the contamination remains unknown.

The Applicant initiated actions to pump out the Unit 1 Containment Spray Sump through a filter/demineralizer system, designed to remove Strontium 90, and investigate the source and means of the Strontium 90 groundwater contamination. This raises the question: is Entergy in violation of the terms of their SAFESTOR for Indian Point 1. When the Applicant

started to remove the underground leaks by pumping the radioactive contamination out of the ground, it caused more radioactive material to be released. Therefore, the NRC ordered the Applicant to stop removing the radioactive effluent from ground, and to monitor it while the issue was further investigated. The NRC has ordered that the contaminated materials remain under the plant in the bedrock, until some date uncertain when Applicant figures out a method to find, stop and remediate the Radiation Leaks. Until that time radioactivity will continue to leach into the groundwater and the Hudson River.

At a recent annual assessment NRC meeting in Croton, NY, NRC officials stated that since they can't dig the radioactive contamination out, and can't blast it out, therefore they will have to chisel out the tritium, cesium and strontium from the bedrock. If such remediation work is required to bring the reactor site into compliance with NRC guidelines and PART 50.7 it will require additional protective actions during the remediation work to keep radioactive contaminants from migrating off site, and exposing both humans, workers and the public, as well as the environment, to unnecessary additional exposure risks and pathways.

In the NRC's _Liquid Radiation Release Lessons Learned Task Force Final Report, ML062650312 2006-09-013.4.3, it was concluded and

recommended that, in some cases, such as Indian Point, the relatively large volumes of contamination above the decommissioning release limits resulted in notable increases in remediation time and costs. The NRC staff estimates the increased cost to be in the tens of millions of dollars, although specific actual cost data is not available to the staff.

The decommissioning reports for Indian Point 2 from 2002 to 2006 indicate that the Urban Inflation rate has been 2.9% per year, yet the adjustment of the decommissioning funds for IP2 has only been 1% per year. However, the decommissioning reports falsely state the escalation rate is 3.0%. The decommissioning funds for Indian Point have a substantial shortfall, as they are not even keeping up with the rate of inflation, as evidenced in the March 29, 2005 Report BVY-05-033/NL-05-039/JNP-05-005/Entergy Nuclear Operations Ltr.2.05.023 and the March 29, 2007 Report Entergy Nuclear Operations C-07-00007.

In addition, the storage of an additional 20 years of waste, either in the spent fuel pools or in dry cask storage, increases the risk to human health and safety far beyond the original Design Basis for this site. Additionally, the NRC has been discussing plans to store both LLRW and HLRW on site at reactor facilities for a period in excess of 100 years, while failing to provide the public with the protection standards that would be in place if a long term

LLRW or HLRW storage facility were cited at the facility. This lack of protections associated with forced onsite storage of radioactive waste streams must be addressed in the license renewal process. Spent fuel pools are not designed to meet the basic minimum requirements for structural stability and integrity, as is outlined in the citing criteria for new reactors in place at the time the NRC granted the original license, and it thus becomes imperative that the structural degradation indicated by the leaks of both Spent Fuel Pools 1 and 2 be addressed and remediated before the license renewal application is allowed to move forward.

Moreover, the dry cask storage facility at Indian Point presents an additional hazard and risk to New York (and other Northeastern states) that will very possibly continue for centuries. The costs of assuming these burdens cannot be placed on the taxpayers, but should be assumed by the Applicant which profits from the operation. These additional costs must be added to the decommissioning fund.

Even the Nuclear Energy Institute (NEI) recommends that although NRC regulations do not require the inclusion of used-fuel storage costs in decommissioning funds, companies should include such costs in their estimates, because no federal repository or interim storage facility is available.

The amount of decommissioning funds required to properly meet the requirements of the NRC 10CFR50.75 are a material issue of fact and law, and a full hearing on such costs and decommissioning funds must occur prior to the NRC approving a new superseding license for 20 years for IP2. The Stakeholders have raised a material matter of fact or law, thus meeting the burden for further review.

5. Contention is Supported by Facts and/or Expert Opinion

Stakeholders have met the minimal requirements of the 10 CFR rules and regulations in presenting this contention in a concise statement of the facts adequate to establish that said contention is entitled to a further and complete review of the issues contained herein. It is pointed out that the rules governing the license renewal process and hearings lay out some basic criteria that a stakeholder must meet to have a contention accepted for further review. Section 2.309(f)(v) requires :

> ....a concise statement of the alleged facts or expert opinion which support the petitioner's position on the issue and on which the petitioner intends to rely at hearing, together with references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue.

Additionally, it is pointed out that the rules and regulations dealing with hearings and contentions accepted therein goes further to define specifically

the minimum burden of proof necessary to have a contention accepted for further review and scrutiny:

> An Intervener is not required to prove its case at the contention filing stage: "the factual support necessary to show that a genuine dispute exists need not be in affidavit or formal evidentiary form and need not be of the quality as that is necessary to withstand a summary disposition motion." Statement of Policy on Conduct of Adjudicatory Proceedings, 48 N.R.C. 18, 22 n.1 (1998), citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process, Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989). Rather, petitioner must make "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate." *In Gulf States Utilities Co.,* 40 NRC 43, 51 (1994), citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process, Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989).

The contention more than meets the minimal standards necessary for acceptance. The petitioner in this case had made "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate."

The Stakeholders assert that the NRC must deny Indian Point 2's LRA because it does not clearly define and allocate decommissioning funds for an aging management program with regard to the adequacy of decommissioning funds or methodology of decommissioning, in light of the underground radioactive leaks, the addition of dry cask storage on site, and the addition of

low-level radioactive waste storage on site, due to the fact that Barnwell is closing in 2008, which by inference affects the LLC's budget for renewal and superseding license within section 50.13 "completeness and accuracy of the information", as it affects the continued aging and safe operations of Indian Point 2.

C. Contention 10: Inability to Access Proprietary Documents Impedes Adequate Review of Entergy Application for License Renewal of IP2 LLC and IP3 LLC.

Issue Statement: FUSE asserts that the Applicant's claims of proprietary status to nuclear industry documents and pertinent sections of the LRA, as well as relevant leak maps and leak reports thwarts the Stakeholders' ability to prepare and file contentions which must be supported by documentary evidence.

Stakeholders that may be adversely affected by a License Renewal Application (LRA) have a right to file a Petition to Intervene and a Formal Request for Hearing. (*§ 2.309 Hearing requests*) There are specific 10 CFR Rules and Regulations that define and spell out the duties and responsibilities of a citizen wishing to use its right to formally intervene in the process, and primary among these rules and regulations, is the filing of a contention. These contentions to be accepted must meet a minimal standard of proof in raising a contention of law or fact which is supported by a methodical presentation of

documents or expert witness testimony in support of the contention. In short, unlike an allegation, contentions must have some supportive evidence that there exists a true difference of opinion of fact or law that falls within the scope of the LRA.

From the date of acceptance of a LRA for review as is witnessed by notice in the Federal Registry, interested Stakeholders usually have exactly 60 days to submit their contentions (on October 2, 2007) with proper evidence, and formally request a hearing and status as an intervener. Stakeholders petitioned the NRC for extension of time to file contentions, and on September 18, 2007, the NRC granted a 60 day extension, until November 30, 2007.

Despite the additional 60 days, the NRC's liberal granting of proprietary status to nuclear industry documents and portions therein, including massive redactions [on the claim of proprietary information] in Application's LRA's for IP2 LLC and IP3 LLC and underlying supporting documents, make it impossible for Stakeholders to adequately review the LRA documents and form/support their contentions. The time necessary to file FOIA's, and to contest the Applicant's claim to proprietary entitlement in keeping documents from public view, or having portions of the LRA and underlying documents redacted takes longer than the time allotted for

Stakeholders to prepare and support their contentions in a fashion adequate to have them accepted for further comprehensive review.

Documents hidden under the guise of proprietary information from Stakeholders are denying Stakeholders their rights to redress under the laws of the United States of America, and under the guidelines of the NRC 10CFR Code of Regulations meant to protect human health and safety.

The time clock for submission of a Formal Request for Hearing, and Petition to Intervene should not begin until stakeholders have access to a full and complete set of un-redacted versions of the LRA and its underlying documents, including but not limited to the FSAR's (all versions), USFAR's (all versions), the most current and up to date company and/or NRC version of the Current Licensing Basis (CLB) which is described in 10 CRF 54.3 as:

> Current licensing basis (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant specific design-basis information defined in 10 CFR 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC

> bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

In addition Stakeholders must be given access to the plume maps and leak reports prepared by the Applicant and exhibited at public meetings, yet claimed as proprietary by the Applicant.

The NRC must use its discretion to weigh the value of public health and safety against commercial interests, especially with regard to environmental information. Until such relevant documents are made available for Stakeholder's review, it is inequitable for the NRC to close the window in which Intervenors may submit contentions and request a hearing with regard to the LRA.

1. Basis for Contention

(i) As Stakeholders and property owners living within 3, 10 or 50 miles of the Indian Point facility owned by two unique and separately owned Entergy Limited Liability Corporations, it is imperative in measuring any suspected and/or adverse risks/effects associated with the proposed actions sought in Entergy's LRA for IP2 LLC and IP3 LLC to have a clear understanding of what the License Renewal Application seeks, and be capable of measuring the reliability and adequacy of the aging management plans contained therein.

(ii) In measuring the potential risks and/or adverse effects associated with the proposed action (license renewal) the Stakeholders have done due diligence in working their way through the myriad complexities in the Applicant's LRA for IP2 LLC. Citizen volunteers, FUSE USA staff, attorneys and our industry expert have dedicated thousands of man/woman hours to fully understanding the repercussions of a 20 year license renewal on the community surrounding Entergy's Indian Point.

Despite best efforts on the part of the Stakeholders, the Applicant's claims of entitlement to Proprietary Information, and the NRC's granting of the Applicant's request for same, has created a situation where Stakeholders are barred from properly forming and supporting certain contentions Stakeholder's chose to raise during the limited window for submission of Intervener Petitions.

(iii) One example of the problems created by the Applicant's Claim of the information being proprietary in nature can be found in a cursory review of the most recent UFSAR's for IP2 LLC and IP3 LLC. The Applicant in its LRA refers to the safety analysis in these documents in justifying many aspects of the aging management program, or lack thereof, that will be relied upon in the 20 year period of operation should their LRA be granted. The redacted and publicly available versions of the USFAR's for IP2

and IP3 have over 80 percent of Chapter 14, which is the Safety Analysis, redacted. If Stakeholders cannot review the Applicant's safety analysis, they cannot formulate opinions based upon the facts or on the adequacy of the Applicant's proposed Aging Management Plan as outlined in the LRA.

Further examples revolve around industry documents that the Applicant relies upon in the formulation of its Aging Management Plans (and defense of same) that are not available for review under the same proprietary claims. Stakeholders know of issues regarding Boraflex degradation/failure in the spent fuel pools which brings into question the reliability and workability of the Applicant's aging management plan for the spent fuel pools at IP2 and IP3. An industry investigation into this issue, and the EPRI report on the findings is not publicly available, and is classified as proprietary in nature, even though taxpayer funds (provided by DOE) were used in the creation of said work product. A challenge to this proprietary claim could take months, even years to resolve.

(iv) One FOIA filed with the DOE has been fulfilled in part, with additional document delivery promised by the DOE, if possible, by October 27, 2007 which is 26 days after NRC's initial deadline and possibly a month before the newly extended deadline, for the filing of contentions. The reason for this delay is that the documents must first be reviewed for proprietary

information by EPRI, and if necessary partially redacted before being made available. Including a copy of the letter from the DOE, Sherwood Martinelli, Vice President of FUSE USA, formally requested that NRC grant an extension of time to file a Formal Request for a Hearing and Petition to Intervene (with contentions). The request for an extension of time to file asked for 60 days from the date the DOE fulfils its commitments under the Federal FOIA guidelines. See Exhibit DD. No official action has been taken on the part of the NRC in even acknowledging Stakeholders' specific request.

(iv) It is a reasonable expectation and contention that the citizens and Stakeholders have fair and adequate access to records and documents that are being used in presenting and justifying the important issues found in the Applicant's LRA for IP2 LLC and IP3 LLC. Until such time as Stakeholders are given adequate access to all relevant documents necessary to perform a full and complete review of the LRA, Stakeholders are being unfairly barred from being able to adequately formulate, create and support viable contentions on issues that directly affect public health and safety.

2. The Contention is Within Scope in the License Renewal Process for Entergy's LRA for IP2 LLC and IP3 LLC

Safety analysis and aging management go to the core of any LRA submitted to the NRC. The ability of Stakeholders to investigate and understand the reliability and quality of the Applicant's Safety analysis

assumptions/claims, and evaluate the reliability of the Applicant's proposed aging management plans for the 20 year period of additional operation are crucial for adequate public involvement in the License Renewal Application Process, and should not be mitigated or minimized in the name of expediting the process, or in the name of the NRC calendar. The current licenses for IP2 and IP3 do not expire until 2013 and 2015 respectively, which means granting an extension of time to file formal requests for a hearing and petitions to intervene (with contentions) until all relevant documents are made publicly available, would not negatively impact either the NRC or their licensee in any meaningful fashion. Conversely, denying a reasonable request for an extension of time to file that would allow the Stakeholders an adequate chance to resolve issues surrounding industry and the Applicant's claims to proprietary privilege will cause irreversible harm to the Stakeholders and the Stakeholders' community.

A community and its citizens' right to be involved in the licensing process is not only in scope, but codified into the 10 CFR rules and regulations that govern the re-licensing process. Further, Stakeholder rights to redress are protected and preserved under the First Amendment of the Bill of Rights, and cannot be marginalized in the name of the Applicant or for the convenience of the NRC.

3. Contention Raises Both Material Issues of Fact and Law

The Constitution and the Bill of Rights ascertain fairness of any rules or regulations promulgated under the authority granted an agency such as the NRC, by the Congress of the United States of America. Specifically, we must look at the First Amendment which states:

> Congress shall make no law respecting an establishment of religion, or prohibiting the free exercise thereof; or abridging the freedom of speech, or of the press; or the right of the people peaceably to assemble, and to petition the government for a redress of grievances.

NRC's authority to promulgate and enforce rules and regulations stems by proxy from a direct act of the Congress of the United States of America. Since the Constitution and Bill of Rights preclude Congress from making laws which abridge the people's right to peaceably petition the government for redress of grievances, the NRC that was created by Congress cannot legally exist, create, draft or enforce any rule or regulation that de facto abridges the people's right to a adequate redress of grievances.

The very nature of the NRC's relicensing rules and regulations as codified in 10 CFR, specifically as they relate to what is and is not within scope, what the NRC allows the Applicant to claim as proprietary, and the limited time allotted for citizens to adequately address and submit their

contentions de facto abridges the people's right to petition the government for a redress of grievances.

There are numerous laws drafted by Congress which show their intent to preserve the individual rights of citizens at all costs against unfair, unjust and illegal ordinances and regulations.

See 42 U.S.C. § 1983 it is, in relevant part, as follows:

> Every person who, under color of any statute, ordinance, regulation, custom, or usage, of any State . . . subjects, or causes to be subjected, any citizen of the United States or other person within the jurisdiction thereof to the deprivation of any rights, privileges, or immunities secured by the Constitution and laws, shall be liable to the party injured in an action at law, suit in equity, or other proper proceeding for redress.

In invoking § 1979 as revised in 42 U.S.C. § 1983 Stakeholders contend that their protection of "rights, privileges, or immunities secured by the Constitution" encompasses what "due process of law" and "the equal protection of the laws" of the First Amendment guarantee against action by the NRC. The withholding by the Applicant of "proprietary documents", such as the leak report and leak plume maps, during the limited time in which the public is permitted to file Formal Request for Hearing, and Petition to Intervene with contentions, deprives and denies Stakeholders their Constitutional rights, and is unduly prejudiced in favor of the Applicant. It is

a blatant attempt by Entergy, NEI and the NRC to deprive Stakeholders of or marginalize the Stakeholders rights and privileges secured by the Constitution.

The NRC, in their method of conducting a License Renewal Process, has deliberately designed it with the assistance of the Nuclear Energy Institute (NEI), the powerful nuclear industry lobbying group, to eliminate any meaningful citizen involvement, and has intended to thwart all chance of real redress, as is guaranteed by the Constitution and Bill of Rights.

Moreover, the Applicant's hiding of crucial documents behind the veil of Proprietary Privilege, and the NRC's granting of privilege without question so that Stakeholders might deal with the legal roadblocks presented by the Applicant's claim of Proprietary Privilege are the very acts that 42 U.S.C. § 1983 was meant to protect against.

The Applicant has deliberately and knowingly caused another person (NRC Staff) to hide and/or withhold documents from and official proceeding (License Renewal Application Process). The Applicant's wrongful and abusive claim to and use of Proprietary Privilege is targeted at thwarting adequate participation by Stakeholders in the official proceeding of the License Renewal Application process, and official proceeding of the Nuclear Regulatory Commission, an agency of the government of the United States of America. NRC's blind granting of said privilege without question of its

licensee's entitlement to same makes both parties guilty of an attempt to withhold and/or alter documents meant for use in an official proceeding, and prejudices the LRA proceedings in favor of the Applicant:

> 18 U. S. C. §§1512(b)(2)(A) and (B) makes it a serious crime to "knowingly ... corruptly persuade another person ... with intent to ... cause" that person to "withhold" documents from, or "alter" documents for use in, an "official proceeding."

Further, the NRC has to weigh a licensee's claim of Proprietary Privilege against the publics'' need to know. It is imperative in making a decision to grant a request for Proprietary Privilege against the right of the public to be fully apprised of the bases for, and the potential effects, risks and health concerns associated with the proposed action .

(i) § 2.390 Public inspections, exemptions, requests for withholding

See subsection B (5) (6)

> (5) If the Commission determines, under paragraph (b)(4) of this section, that the record or document contains trade secrets or privileged or confidential commercial or financial information, the Commission will then determine whether the right of the public to be fully apprised as to the bases for and effects of the proposed action outweighs the demonstrated concern for protection of a competitive position, and whether the information should be withheld from public disclosure under this paragraph. If the record or document for which withholding is sought is deemed by the Commission to be irrelevant or unnecessary to the performance of its functions, it will be returned to the applicant.

> (6)Withholding from public inspection does not affect the right, if any, of persons properly and directly concerned to inspect the document. Either before a decision of the Commission on the matter of whether the information should be made publicly available or after a decision has been made that the information should be withheld from public disclosure, the Commission may require information claimed to be a trade secret or privileged or confidential commercial or financial information to be subject to inspection under a protective agreement by contractor personnel or government officials other than NRC officials, by the presiding officer in a proceeding, and under protective order by the parties to a proceeding. In camera sessions of hearings may be held when the information sought to be withheld is produced or offered in evidence. If the Commission subsequently determines that the information should be disclosed, the information and the transcript of such in camera session will be made publicly available.

From the onslaught of the Applicant's LRA process for IP2 LLC, the NRC failed in their fiduciary duties and responsibilities to members of the public when it comes to the Applicant's claims of Proprietary Privilege. Instead of making a decision based on the public's need to know weighed against Entergy's desire to hold a competitive edge in the nuclear industry, the NRC staff, as a matter of practice, simply grants all requests by the Applicant for Proprietary Privilege. NRC's in-house protocols, in this regard, fly in the face of its own regulations, and have placed members of the public at a grave disadvantage by interfering with the Stakeholders' rights to redress

in this case, and by interfering with Stakeholders' ability to file properly supported contentions.

The Stakeholders of the host community are being told by both Applicant and NRC to simply trust them. Past review of Indian Point, and the regulatory problems associated with the site, show there is reason not to trust either the licensees or the NRC. As President Ronald Reagan said, "trust but verify." Stakeholders cannot verify, cannot ascertain the accuracy of the Applicant's Safety Analysis, nor can Stakeholders accept the Applicant's proposed aging management analysis without a full review of the application and its underlying documents. Further, and germane to this contention, Stakeholders cannot adequately and timely prepare properly prepared and researched contentions without unfettered access to the full record in this case.

It is clear, that in the case of the Applicant's LRA for two aging reactors with known Flow Accelerated Corrosion (FAC) issues, known fatigue issues, known cross-cutting issues, and a host of other safety and equipment failures that the public's right and need to know should outweigh Entergy's need for secrecy, should outweigh the NRC's desire to keep to a tight time schedule in the relicensing process.

4. Contention is Supported By Facts and/or Expert Opinion

Intervener has met the minimal requirements of the 10 CFR rules and regulations in presenting this contention in a concise statement of the facts adequate to establish that said contention is entitled to a further and complete review of the issues contained herein. It is pointed out that the rules governing the license renewal process and hearings lay out some basic criteria that a stakeholder must meet to have a contention accepted for further review. *Section 2.309(f)(v) requires,*

> ...a concise statement of the alleged facts or expert opinion which support the petitioner's position on the issue and on which the petitioner intends to rely at hearing, together with references to the specific sources and documents on which the petitioner intends to rely to support its position on the issue.

The contention as written clearly presents a concise statement of the alleged facts and matters of law. Without first resolving the matters surrounding production of documents, without first reaching agreement on what documents are or are not entitled to Proprietary Privilege, it is impossible for interveners to adequately review Entergy's LRA in a meaningful fashion and submit our contentions in a timely fashion.

The right to add additional supporting documents, and name industry expert witnesses and the scope of their testimony is fully reserved herein. Additionally, it is pointed out that the rules and regulations dealing with hearings and contentions accepted therein goes further to define specifically

the minimum burden of proof necessary to have a contention accepted for further review and scrutiny:

> An Intervener is not required to prove its case at the contention filing stage: "the factual support necessary to show that a genuine dispute exists need not be in affidavit or formal evidentiary form and need not be of the quality as that is necessary to withstand a summary disposition motion." Statement of Policy on Conduct of Adjudicatory Proceedings, 48 N.R.C. 18, 22 n.1 (1998), citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process, Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989). Rather, petitioner must make "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate." In Gulf States Utilities Co., 40 NRC 43, 51 (1994), citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process, Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989).

It is clear here, that this contention more than meets the minimal standards necessary for acceptance of this contention. The petitioner in this case has made "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate."

(i) Contention Raises a Material Matter of Fact or Law

The adequacy of a 60 day time period from the date of acceptance of Entergy's LRA as witnessed by notice of same in the Federal Registry is by fact subjective, and up to interpretation. The effect that the Applicant's claim of Proprietary Privilege has on the Stakeholder community's ability to disseminate and understand the LRA and submit properly supported

contentions in a timely fashion is also a subjective issue of fact that should be decided by an impartial board or in a court of law. The Applicant's entitlement to its claim of Proprietary Privilege is, or should be subjective in scope. The constraints and limitations the NRC's time constraints have placed on our community's right to redress and limited by the Applicant's claim to relevant documents as "proprietary" is a matter of law in dispute, and should also be resolved by a board or in a court of law.

E. Contention 11: Regulatory Guidance contained in 10 CFR50.4 and Rule Implementing Standards under the American Rules and Procedures Act require Stakeholders to have reasonable opportunity to bring forth issues beyond the narrow scope where members of the public have specific and direct substantiated concerns

Issue Statement: Stakeholders assert that 10 CFR50.4 and Rule Implementing Standards under the American Rules and Procedures Act, require Stakeholders to have reasonable opportunity to bring forth issues beyond the narrow scope so that members of the public can raise specific and directly substantiated concerns, including but not limited to, an Independent Safety Assessment.

Regulatory Guidance contained in 10 CFR50.4 and Rule Implementing Standards under the American Rules and Procedures Act require Stakeholders to have reasonable opportunity to bring forth issues beyond the narrow scope where members of the public have specific and directly substantiated

concerns. The Stakeholders and elected officials (including, Senator Hillary Clinton, Governor Eliot Spitzer, Congresswoman Nita Lowey, Congressman John Hall, Congressman Eliot Engel, Congressman Maurice Hinchey, as well as Westchester, Rockland, Putnam and Dutchess Counties Executive and Legislators, and the municipalities of Village of Croton-on the Hudson, City of Beacon, Village of Ossining, Town of Cortlandt, Town of Ramapo, Town of Stony Point, and Town of Putnam Valley) call for an Independent Safety Assessment (ISA) of Indian Point systems, components and programs beyond the narrow recommendations of existing regulatory guidance. NRC's denial of this Independent Safety Assessment and the NRC's current mode of oversight increasingly reduces accountability and transparency. Stakeholders assert that these issues must be fully addressed and resolved prior to final license renewal. These areas of scope include 4.16 KV electrical distribution system, Control Ventilation, containment ventilation, and many more issues. (See exhibit AA)

F. Contention 12: The LRA, in which Indian Point 2 LLC seeks a new superseding license to replace the existing license, is incomplete and should be dismissed, because instead of presenting required Time Limiting Aging Analysis and an Adequate Aging Management Plan, it seeks to agree to uncertain commitments with regard to the Aging Management of the plant at an uncertain date in the future, thereby causing the license agreement to be voidable.

The Stakeholder's contend that the Applicant has submitted an LRA that contains uncertain and undefined commitments with regard to its Aging Management Plan, and therefore cannot be approved by the NRC because it is non-binding and is merely an intention to "agree to agree" to a plan that will be defined sometime in the future. Instead of presenting specific plans required for Time Limiting Aging Analysis (TLAA) and adequate aging management plans to deal with known plant degradation issues, the proposed LRA merely provides commitments in the licensing review process to conduct certain Time Limiting Aging Analysis (TLAA), and implement as yet unknown Aging Management Plans at some future date and time. The NRC's job is to identify shortcomings in the application and identify unaddressed issues in the application, not to negotiate with the Applicant in the review process for a list of future non descript commitments. A TLAA either was done to address a known aging issue, or it was not. An aging management plan either exists, or it does not. If it does not exist, if the analysis has not been done, the application is incomplete. A future commitment to complete a TLAA amounts to nothing more than an agreement to agree to an analysis that has not yet been completed, and therefore an Aging Management plan cannot be developed and/or committed to, until an uncertain date in the future, thereby making the terms of the license vague, non-specific and unenforceable.

NRC's 10 CFR 54 in part requires a licensee to A) conduct a Time Limiting Aging Analysis (TLAA) for primary equipment and components subject to fatigue that are determined to be in scope, and B) require as a part of the license renewal application that adequate Aging Management Plans be included in the application to deal with any parts, components, and systems that are subject to aging issues such as fatigue that are within scope.

Said regulations deliberately do not provide a mechanism for a plan to be submitted at some later date. Moreover, allowing such a future commitment not only bars public Stakeholder involvement in the process, thereby removing the review of said aging management plans from public scrutiny, it also violates the intent of the regulations, if not the regulation itself. The LRA is supposed to be complete, and address **all issues** involved in licensee being granted a new superseding license. Making a commitment to address the issues of an Aging Management Plan later on is not the intent of the law. Agree to agree is not law.

Further, the NRC is now realizing that many previous licensees who have moved through the re-licensing process are finding it impossible to meet the deadlines set for those future commitments. Even more disturbing is that the NRC is discussing the possibility of granting these licensees relief from

those very commitments, in a classic example of "out of sight out of mind." This process needs to be transparent, and the NRC needs to act as a regulator who abides by and enforces its rules and regulations, rather than acting as an arbiter and deal maker. The License Renewal Process is a serious and regimented process, not "Let's Make A Deal".

The NRC has, in past LRA proceedings allowed the Applicant to make a future commitment to A) perform an assessment of this known fatigue issue, and B) make a future commitment on the part of the Applicant to devise an acceptable aging management program for this known issue at some later date after the license renewal application has been approved.

The thousands of letters of relief from NRC rules, and licensee commitments show that this is not acceptable. As an example, it is pointed out that Indian Point 2 made a commitment when first licensed back in the early 70's to design and build a **closed cooling system**. Some 30 plus years later, Entergy is still rationalizing the missed commitment that originally had a 1979 date of delivery.

The Stakeholders in the current LRA proceeding regarding IP2 contend this method is unacceptable and makes the license unenforceable. The 10 CFR rules are very specific and include the language without ambiguity that "licensees are to have an aging management plan in place for

review". Agreeing to keep an eye on things while you invent/create an aging managing plan does not meet the regulations as they now are written and exist. Agreement to agree is not legally enforceable under basic contract law.

In the current LRA proceeding and approval process the Applicant makes a commitment to the NRC to vaguely do something left basically undefined at some uncertain future date and time after a new superceding license has already been issued. This amounts to nothing more than an agreement to agree later on a process that remains, at best, vaguely defined.

In order to be a valid and enforceable agreement, a document must contain certain essential legal provisions and must not leave either undecided or to be determined at some time in the future any aspect of such essential legal provisions. If these essential elements are not present, then the document is not a binding one and is often referred to by courts as an "agreement to agree" and is nothing more than a letter of intent, both of which are not enforceable as contracts or license. A license is essentially a contract between a regulator and a regulated business, in this case the NRC and IP2 LLC.

In *Richie Co. LLP vs. Lyndon Insurance Group*, Inc., a federal case out of the Eighth Circuit interpreting Minnesota law, the Court held that the April

16, 1999 "agreement" was not an agreement at all but a non-binding letter of intent and agreement to agree. The Court stated: A letter creating an agreement to negotiate in good faith in the future is not enforceable where the parties have contemplated that the agreement is not the complete and final agreement governing the transaction at issue.

The Court also stated: Furthermore, where the parties have agreed that an "agreement to negotiate" or letter of intent, in its entirety, is not a binding legal agreement, Courts have refused to enforce an individual provision of the letter as a freestanding "contract" promise. Therefore since the NRC plans to accept vague commitments with unspecified protocols to be determined at an uncertain date in the future, for certain components and systems in IP2's aging management plan, then the entire plan and new superseding license will be unenforceable and void.

The Court stated further: That language that spoke of future actions and agreements contemplated but not yet completed by the parties showed that the letter "was not the complete and final agreement the parties contemplated would govern" but "merely created an agreement to negotiate in good faith." Such language clearly manifests an intention to do something essential at a later date, thus the document is not a binding contract but merely an unenforceable agreement to agree and a non-binding letter of intent. A

nuclear reactor applicant must not be allowed to operate a facility without a complete and fully enforceable legal license with specific terms of the license in place.

If the NRC approves a new superseding license based on the Applicant's LRA that contains criteria and obligations of the Applicant that do not have sufficient certainty with regard to the aging management plan, then such License will be void for uncertainty. The new superseding license will be nothing more than an "agreement to agree", as to essential terms and conditions, that may adversely affect public health and safety left vague and uncertain, to be defined at an uncertain date and time.

In addition the NRC's acceptance of the Applicant's proposed LRA with uncertain and vague criteria, will bar Stakeholders from participating in the review of specific criteria that may adversely affect public health and safety, which is a violation of Stakeholders right to both due process and full redress under the law.

Therefore the Stakeholders contend that the NRC cannot approve the LRA with any vague or uncertain criteria, with unenforceable future commitments which would cause the new superseding license to be unenforceable and void.

G. Contention 13: The LRA submitted fails to include Final License Renewal Interim Staff Guidance. For example, LR-ISG 2006-03, " Staff guidance for preparing Severe Accident Mitigation Alternatives."

The LRA submitted fails to include Final License Renewal Interim Staff Guidance (LR-ISG) For example, LR-ISG 2006-03, " Staff guidance for preparing Severe Accident Mitigation Alternatives (SAMA)." This License Renewal Interim Staff Guidance recommends that applicants for license renewal use the Guidance Document Nuclear Energy Institute 05-01, Revision A, (ADAMS Accession No. ML060530203) when preparing SAMA analyses. The Applicant failed to include any Interim Staff Guidance in its submittal in spite of the recommendation of the NRC, and in spite of the regulator incorporating License Renewal Interim Staff Guidance in the next revision of Supplement 1 to Regulatory Guide 4.2, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses." Here the Applicant failed to address not just the rule but failed to address the trade guidance documents as well. (see Exhibit BB)

H. Contention 14: The Updated Final Safety Analysis Report (UFSAR) fails to meet the requirements of 10 C.F.R.55(a) by deletion of required codes and standards, and obviates the ability for a petitioner to perform a technical review as required under 10 CFR 50.4.

Statement of Issue: The Stakeholders assert that The Updated Final Safety Analysis Report (UFSAR) as referenced in the LRA for Unit 2 fails to meet the minimum requirements of 10 CFR 55(a), and fails to include codes and standards required to be contained in the UFSAR. This fundamental and cornerstone document was altered between the years 2000, and 2006 to remove essentially all codes and standards and therefore is prima facie in violation of federal rules. Without the Safety Analysis Report including necessary codes and standards the license to operate the facility has no basis to ensure the safe operation and protection of the health and safety of the public.

The Updated Final Safety Analysis Report (UFSAR) as referenced in the LRA for Unit 2 fails to meet the minimum requirements of 10 CFR 55(a), and fails to include codes and standards required to be contained in the UFSAR. This fundamental and cornerstone document was apparently altered between the years 2000, and 2006 to remove essentially all codes and standards and therefore is prima facie in violation of federal rules. Without the Safety Analysis Report including necessary codes and standards the license to operate the facility has no basis to ensure the safe operation and protection of the health and safety of the public.

I. Contention 15: The Applicant does not have in its possession the Current License Basis (CLB) for Indian Point 2, that is required for license renewal under CFR 2.390

Statement of the issue: FUSE asserts that the Current License Basis for Indian Point 2 is unknown and unavailable, thereby preventing the right of Stakeholders the right to review and analyze plant specific commitments and modifications.

The Applicant is required to have in its possession and control the precise current license basis for each unit. The current license basis (CLB) is defined in 10CFR50.3. The Current License Basis (CLB) is required for Renewal as required for license renewal[3] under the CFR 2.390. is unavailable.

Numerous attempts have been made by the NRC as well as the GAO to determine whether the Current License Basis is known, current, documented, and available. None have been successful. The most recent was an investigation by the GAO (See exhibit X), where it was concluded that the CLB for each plant is ***not known***. This is particularly material, given that the

[3] Of note is that very recently numerous examples of non existent CLB were requested and denied, the licensee or the regulatory agency have begun to address parts of this issue, for example, the General Design Criteria were made available after numerous requests but only recently. The same for the SERs and FSARs (but on heavily redacted form). The CLB by rule under CFR54.3 is plainly interpreted that the pertinent parts must be available at the beginning of the public review period, and not two weeks before the end of the 60 day review window. See exhibit xxx for detailed correspondence history regarding some of these documents, letters, rejections, and emails. Even with the extension granted under FUSE request for an additional 60 days, this issue still stands.

pertinent parts of the CLB are required under §2.309 to be available to Stakeholders regarding the license renewal of the plant.

The CLB includes the Design Basis Document Program. For IP3 this is referred to as the Design Basis Verification Program, for Indian Point 2, this is referred to as the Design Basis Document Program. The status of design basis program is outdated, and is not reliable as design basis documents. See for example the IP2, IP3 DVP document regarding Appendix R and Fire Protection. These documents are part of the licensing basis and must have current and relevant portions available to interested parties.

The time clock for submission of a Formal Request for Hearing, and Petition to Intervene should not begin until Stakeholders have access to a full and complete set of un-redacted versions of the LRA and its underlying documents, including but not limited to the FSAR's (all versions), USFAR's (all versions), the most current and up to date company and/or NRC version of the Current Licensing Basis (CLB) which is described in 10 CRF 54.3 as:

> Current licensing basis (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant specific design-

basis information defined in 10 CFR 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

FUSE takes the position that any referenced documents associated with the above is also part of the licensing basis are incorporated by reference into the LRA. Further, it is at the Stakeholder's discretion to determine which of those references are pertinent to performing an adequate technical review of the LRA submitted by the Applicant.

Therefore the NRC must deny the Applicants LRA because the Current License Basis (CLB) required for license renewal under 10CFR2.336 is unavailable and unknown.

J. Contention 16: The Applicant's claims of entitlement to Proprietary Information, and the NRC's granting of their request for same have forestalled petitioners capability of properly forming and supporting certain contentions we wish to raise in the 60 day limited window of opportunity being given by the NRC.

Statement of Issue: Stakeholder's content that despite the notice of extension of September 18 granting a partial extension of the deadline to November 30, 2007, the NRC has not responded specifically and directly to

FUSE Vice President Sherwood Martinelli's request for and extension of time to file asked for 60 days from the date the DOE fulfils its commitments under the Federal FOIA guidelines to provide requested documents on or about October 27, 2007.

An example of the problems created by Applicants' claim of the information being proprietary in nature can be found in a cursory review of the most recent UFSAR's for IP2 LLC and IP3 LLC. Applicant in their LRA refer to the safety analysis in these documents in justifying many aspects of the aging management program (or lack there of) that will be relied upon in the 20 year period of operation should their LRA be granted. The redacted and publicly available versions of the USFAR's for IP2 and IP3 have over 80 percent of Chapter 14 which is the Safety Analysis has been redacted. If Stakeholders cannot review Applicant's 's safety analysis, we cannot formulate opinions in the absence of facts as to the adequacy of their proposed aging management plan as outlined in the LRA.

Further examples revolve around industry documents that Applicant relies upon in the formulation of their aging management plans (and defense of same) that are not available for review under the same proprietary claims. We know for instance that there are issues regarding Boraflex degradation or actual failure in the spent fuel pools which brings into question the reliability

and workability of Applicant's aging management plan for the spent fuel pools at IP2 (and unit1 and IP3). An industry investigation into this issue, and the EPRI report on the findings is not publicly available, and is classified as proprietary in nature, even though tax payer funds (provided by DOE) were used in the creation of said work product. A challenge to this proprietary claim could take months, even years to resolve.

One FOIA filed with the DOE has been fulfilled in part, with additional document delivery promised by the DOE, if possible, by October 27, 2007 which is 26 days after NRC's deadline for the filing of contentions. The reason for this delay, is that the documents must first be reviewed for proprietary information, and if necessary partially redacted before being made available. Including a copy of the letter from the DOE, Sherwood Martinelli, Vice President of FUSE USA, formally requested that NRC grant and extension of time to file a Formal Request for a Hearing and Petition to Intervene (with contentions). The request for an extension of time to file asked for 60 days from the date the DOE fulfils its commitments under the Federal FOIA guidelines. As of September 18 official notice was provided to FUSE at extensive discussions with FUSE, and the extension was granted in part by extending the deadline to November 30, 2007. It remains a legal issue whether the review period should begin when all the document pertinent

review become available under CFR 2.309 after availability. We therefore consider the contention open, and request that it admitted by the Board regardless of the extension FUSE was successful in obtaining.

K. Contention 17: Safety/Aging Management: Applicant's LRA for Indian Point 2 is insufficient in managing the environmental equipment qualification required by federal rules mandated after Three Mile Island that are required to mitigate numerous design basis accidents to avoid a reactor core melt and to protect the health and safety of the public .

Issue Statement: Stakeholder's contend that Applicant's LRA for Indian Point 2 is insufficient in managing the equipment qualification required by federal rules mandated after Three Mile Island that require the Applicant to mitigate numerous design basis accidents established to avoid a reactor core meltdown and to protect the health and safety of the public, because the proposed LRA is not sufficient to demonstrate compliance with either 10 CFR50.49(e)(5) or 10 CFR54.

Summary of Contention

Indian Point 2's LRA does not adequately address the license renewal requirements of 10CFR54 specifically under 10CFR50.54.4, Scope, for those components required for renewal defined in 10 CFR §50.49(b)(1). Indian Point claims credit in their LRA under Table 3.6.1, and EQ analysis in section 4.4 out of compliance with the Rule:

(i) "EQ equipment is not subject to aging management review because replacement is based on qualified life. EQ analyses are evaluated as TLAAs in Section 4.4.

(ii) The Non-EQ Insulated Cables And Connections Program will manage the effects of aging. This program includes inspection of non-EQ electrical and I&C penetration cables and connections.

(iii) The Non-EQ Instrumentation Circuits Test Review Program will manage the effects of aging. This program includes review of calibration and surveillance testing results of instrumentation circuits"

The proposed programs are not sufficient to demonstrate compliance with either 10 CFR 50.49(e)(5) or with 10 CFR 54.

Essentially, Entergy under the approval of the NRC, but with objection of the Advisory Committee on Reactor Safeguards (ACRS), found alternative analysis that performed a rudimentary economic analysis to disregard federal rules regarding Entergy's current license basis (CLB) with respect to equipment required to operate during a design basis accident. A rudimentary quality study procured by the NRC concluded that a 50 % chance of multiple equipment not functioning was acceptable. based upon an economic analysis.

This flagrant abuse of federal rules, and non-compliance with the Federal Administrative Procedures Act might be compared to a school district deciding to remove all the fire extinguishers in a school district because the chances of a fire are low, and the cost of keeping them in operating condition is high, regardless of a law mandating public schools with 100s of students in attendance to have the extinguishers present, operable and inspected at prescribed times. To illustrate, a high school administrator questioned the need for fire extinguishers because of costs and historical absence of fires, and literally not one extinguisher used. Even though the administration knew that the law required extinguishers to placed, and maintained, it acted negligently by knowingly keeping some brands that may not properly function or simply fail. So instead of fulfilling the legal requirement of having working extinguishers, the school administration deliberately set aside the requirement in lieu of an alternative probabilistic risk analysis (PRA) study – to save money. The fire extinguishers were quietly thrown out as each one broke etc. At Indian Point 2, the Applicant and NRC concluded that the economic analysis to justify a 50% failure rate was acceptable.

1. Applicable Federal rules pertaining to this contention

(i) Under §54.19 of requirements for license renewal, Applicant must provide the information specified in 10CFR50.33(a) through (e), (h)

and (i)...or by reference to other documents that are required for this section. Under §54.21, Contents of the application—technical information, each application must contain the following information:

(A) ***An integrated plant assessment (IPA).***

(1) For those systems, structures, and components within the scope of this part, as delineated in §54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components:

a. That perform an intended function, as described in § 54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air

compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

b. That are not subject to replacement based on a qualified life or specified time period.

(2) Describe and justify the methods used in paragraph (a)(1) of this section.

(3) For each structure and component identified in paragraph (1)(i) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

(B) ***CLB changes during NRC review of the application.*** Each year following submittal of the license renewal application and at least 3 months before scheduled completion of the NRC review, an amendment to the renewal application must be submitted that identifies any change to the CLB of the facility that materially affects the contents of the license renewal application, including the FSAR supplement.

(C) ***An evaluation of time-limited aging analyses.***

(1) A list of time-limited aging analyses, as defined in § 54.3, must be provided. The applicant shall demonstrate that—

a. The analyses remain valid for the period of extended operation;

b. The analyses have been projected to the end of the period of extended operation; or

c. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

(2) A list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses as defined in § 54.3. The applicant shall provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

(D) ***An FSAR supplement.*** The FSAR supplement for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation determined by paragraphs (a) and (c) of this section, respectively.

a) Under License Renewal Rule 10 CFR 54, Entergy must specify components that are within the scope and in particular those that are defined under the requirements of 10 CFR 50.49. 10 CFR § 54.4 Scope specifies that plant systems, structures, and components within the scope of the License Renewal Rule are: Safety-related systems, structure*s, and compon*ents which

are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions:

b) Plant systems, structures, and components within the scope of this part are Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions:

a. The integrity of the reactor coolant pressure boundary;

b. The capability to shut down the reactor and maintain it in a safe shutdown condition; or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.

d. All non-safety related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section..

e. All systems, structures, and components relied on in safety analysis or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

f. The intended functions that these systems, structures, and components must be shown to fulfill in § 54.21 are those functions that are the basis for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this

section.[60 FR 22491, May 8, 1995, as amended at 61 FR 65175, Dec. 11, 1996; 64 FR 72002, Dec. 23, 1999].

2. Analysis of the of Indian Point 2 LRA Against the Rule

(i) The Indian Point application for Unit 2 for License renewal, as it applies to Equipment Qualification Program MUST consider the following requirements of 10CFR 50.49:

(A) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(B) The validity of partial test data in support of the original qualification.

(C) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(D) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(E) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.

(ii) Issues regarding 10 CFR 50.49 were identified under a Generic Safety Issue number 168.

Issues regarding 10 CFR 50.49 were subsequently investigated by numerous parties. Many components were found unqualified to function for the 40 years let alone 60 years. These components are presently installed at

Indian Point 2 and 3. See exhibit FF Certain components and failures were found as high as 50%. ***Id.***

(iii) The Advisory Committee for Regulatory Safeguards (ACRS) reviewed the results of GSI 168 and ACRS Comments on GSI 168, and then made a number of recommendations[4]

A discussion of the treatment of the instrumentation and control (I&C) cables during the license renewal term be included in the generic communication recommended by RIS 2003-09 see exhibit V. The staff encouraged the industry to perform further developmental work on techniques for monitoring Instrumentation &Control (I&C cable) condition. The staff concluded that the current equipment qualification (EQ) process for low-voltage Instrumentation &Control (I&C) cables is adequate for the duration of the current license term of 40 years. Knowledge of the conservatism in the operating environment, as compared to the qualification environment, coupled with observation of the condition of the cables can be used to extend the qualified life of the cables. A combination of condition monitoring techniques is needed since no single technique is effective to detect degradation of Instrumentation &Control (I&C) cables. Test results and other pertinent information should be disseminated to the nuclear industry through a generic communication.

[4] ACRS letter dated June 17, 2002

(iv) Additional Comments by Advisory Committee for Regulatory Safeguards (ACRS) Members Dana A. Powers, F. Peter Ford, Victor H. Ransom, Stephen L. Rosen, and John D. Sieber include the following:

The staff has recommended a resolution of cable integrity issues for one class of design-basis accidents, loss-of-coolant accidents. For these accidents, temperature and radiation loads are of dominant concern. Other design-basis accidents, such as main steam-line breaks, can impose other loads on cables such as large amplitude vibrations and bending. The staff has not investigated the effects of these other loads on the integrity of aged cables adequately. What the staff has done is adequate to resolve the six, open, sub-issues of GSI-168. The staff should consider additional examinations of cable integrity as part of its ongoing work on mechanical loads and vibrations associated with main steam-line breaks and other design-basis accidents.

MR. AGGARWAL: Thank you.

As we reported to you previously, there were failures of certain I&C cables in NRC tests, namely in LOCA test numbers 4, 5, and 6. Failures of single conductor bonded Okonite cables. Sampled more cables in test number 4, and eight out of 12 cables failed in LOCA test number 6 for 60 years. We also found in our research that there is no single condition monitoring technique available which is effective to detect degradation. Probably combination of different techniques can be used, depending upon the type of insulation. We also found that visual inspection can be useful in assessing the degradation of cable with time. (Pg. 224-225)

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numbers 4, 5, and 6. Failures of single conductor bonded Okonite cables. Sampled more cables in test number 4, and eight out of 12 cables failed in LOCA test number 6 for 60 years. We also found in our research that there is no single condition monitoring technique available which is effective to detect degradation. Probably combination of different techniques can be used, depending upon the type of insulation. We also found that visual inspection can be useful in assessing the degradation of cable with time. (Pg. 224-225)

Turning to the 60-year aging assessment, which was LOCA test number 6, in our test, eight out of 12 cables failed the post-LOCA test. And we have concluded that some of these cables may not have sufficient margin beyond the 40 years of the qualified life. (Pg . 233-234)

(v) Brookhaven Testing, 4.5.4 Extending Qualified Life (exhibit FF)

The data obtained from test sequence 6 are of particular interest for the issues related to extending qualified life. In that test, cables from four different manufacturers were pre-aged to the equivalent of 60 years of qualified life and were then exposed to simulated LOCA conditions. As discussed in Section 3.6, a number of the specimens experienced degradation related failures during a submerged voltage withstand test in which they were unable to hold the test voltage. These results indicate that the degradation due to aging beyond the qualified life of the cables may be too severe for the insulation material to withstand and still be able to perform during an accident. For life

extension purposes, the qualified life of the cables should be reviewed and compared to actual plant service environments. A determination can then be made as to whether the additional exposure to aging stressors during the period of extended operation will be acceptable for the cable materials.

(vi) Under RIS 2003-09, The NRC accepted the Advisory Committee for Regulatory Safeguards (ACRS) in part, and set aside significant technical concerns in other parts. This is a clear violation. Five members dissented in accepting the study closing GSI.

The staff has concluded that, although a single reliable condition-monitoring technique does not currently exist, walk downs to look for any visible signs of anomalies attributable to cable aging, coupled with monitoring of operating environments, have proven to be effective and useful.

A combination of condition-monitoring techniques may be needed since no single technique is currently demonstrated to be adequate to detect and locate degradation of Instrumentation &Control (I&C cables). Monitoring Instrumentation &Control (I&C) cable condition could provide the basis for extending cable life.

The apparent violation of the Administrative Procedures Act because of the NRC's bypassing of ACRS recommendations regarding compliance to 10CFR50.49, and the implications to 10CFR50.4.

(vii) Regulatory Issue Summary (RIS 2003-09)

The staff has concluded that, although a single reliable condition-monitoring technique does not currently exist, walk downs to look for any visible signs of anomalies attributable to cable aging, coupled with monitoring of operating environments, have proven to be effective and useful. A combination of condition-monitoring techniques may be needed since no single technique is currently demonstrated to be adequate to detect and locate degradation of Instrumentation &Control (I&C) cables. Monitoring Instrumentation &Control (I&C) cable condition could provide the basis for extending cable life.

Expert Witness testimony

See Declaration by Expert Witness Ulrich Witte, regarding his work with Equipment qualification and Arrhenius aging as was implemented in the 1980s, then questioned regarding license renewal. See exhibit GG

2. **Conclusion**

The NRC violated Title 5, Part I, Chapter 7 of the Federal Administrative Procedures Act—and that the problem has particular relevance to Indian Point 2 license renewal as well as IP2's present ability to cope with certain design basis accidents.

Particularly in 10CFR50.49.

The following are multiple component examples required for safe shutdown of the IP2 –which are presently unqualified and will apparently remain unqualified from Entergy statements in their LRA. FUSE argues (1) the violations made by Entergy in failing to comply with the 10CFR50.49 (2) the violations made by the regulatory agency, the NRC, in accepting the unqualified components as okay, even with a flawed approval based upon industry guidance, that actually violate the law. (3) the NRC recognized its own errors, and in a series of actions beginning about five YEARS AGO deliberately bypassed the Administrative Procedures Act in an attempt to cover up the blunder by using an unlawful procedural process of probabilistic cost analysis (PRA) and cost benefit analysis, thereby dismissing issues with which Advisory Committee for Regulatory Safeguards (ACRS) found fault

The NRC then closed out the issue articulating supposed endorsement from the Advisory Committee for Regulatory Safeguards (ACRS), notwithstanding the Advisory Committee for Regulatory Safeguards (ACRS) stated concerns. (4) The GAO has noticed the approach taken by the NRC and Entergy on other issues, yet Entergy failed to act to comply with the regulations. This was the case, in particular, with respect to Indian Point 2.

The recent documents show the NRC intended to set aside compliance with federal rule 10 CFR 50.49. The Applicant obviously proposes that the present proceedings for relicensing the Indian Point plants yield no alternative other than for the public to accept the violations by Entergy and the NRC—and the consequential unsafe material conditions of the plant to withstand the design basis requirements specified in the current UFSAR, as well as, the proposed amended UFSAR for license renewal. New testing done by laboratories under contract from the NRC show cable failure rates on the order of 50%. Yet they closed the issue regardless under a high school quality economic analysis. The approach was not only unlawful but also, technically irresponsible. Ensuring the functionality of the numerous cables and components required for safe shutdown is one of the major requirements that licensees are required to perform because of the events of Three Mile Island (TMI). Some consider these actions *the* most major. To bypass them now is beyond reason, and violates the NRC's mandate to adequately protect public health and safety.

This contention should be admitted as is a matter of law, and as a matter of fact.

Therefore the NRC must deny the Applicant's LRA because it does not adequately address the license renewal requirements of 10CFR54, specifically

under 10CFR54.4, for those component required for renewal defined in 10CFR50.49(B)(1) for an aging management plan, thereby failing to adequately protect public health and safety.

L. Contention 18: Entergy's License Renewal Application Does Not Include an Adequate Plan to Monitor and Manage Aging of Plant Piping Due to Flow-Accelerated Corrosion During the Period of Extended Operation.

Issue Statement: Stakeholders assert that Indian Point 2 LLC's IP2 LLC's License Renewal Application does not include an adequate plan to monitor and manage aging of plant piping due to Flow-Accelerated Corrosion (FAC), as required pursuant to 10 C.F.R. § 54.21(a)(3). The plant piping is subject to aging management review, pursuant to 10 C.F.R. § 54.21(a), and FAC is an aging phenomenon that must be adequately managed. *See* NUREG-1801, *Generic Aging Lessons Learned (GALL) Report,* Revision 1, U.S. Nuclear Regulatory Commission. FUSE submits the Declaration of Mr. Ulrich Witte in support of this contention.

Scope and approach of the Flow –Accelerated Corrosion is noted as unchanged as compared to the present licensing basis. Therefore, by implication, scope of the program includes:

1. Extraction Steam System: (see e.g. IP3-RPT-EX-0911 for Unit 3)

2. Condensate System: (IP3-RPT-COND-0912)

3. Moisture Separator Drain System: (IP3-RPT-HD-00913)

4. Heater Drain System: (IP3- RPT-HD-00979)

5. Feedwater System: (IP3-RPT-0984)

6. Reheater Drain System: (IP3-RPT-HD-01144

7. Moisture Separator Drain System: (IP3-RPT-MSD-01158)

8. Historical Inspection Data: (IP3-RPT-MULT-01471)

9. Small Bore and Augmented Piping Program: (IP3-00064.000-1)

A review of an Advisory Committee for Regulatory Safeguards (ACRS) Transcript discussing the predictability of the industry accepted technical approach cited by the Applicant is precisely on point and worth of quoting the dialogue directly by the ACRS and the admissions by Entergy regarding the weakness in reliability of the methodology, and specifically addresses the Extraction Steam System. Mr. Rob Alersick of Entergy made the following comments during ACRS 2003 meeting in Rockville Chaired by Dr. GrahamWallis:

> Mr. ALERISK, [Entergy]: I've had the opportunity to be involved with flow accelerated corrosion since 1989 and in particular have modeled or otherwise addressed approximately 20 EPU efforts in the last two years. Dr. Ford made a very good point earlier when he said that the graph that we looked at did not

display a very good correlation between the measured results and the predicted results out of CHECWORKS. Programmatically—well, let me back up a second. That is certainly true in the example that we looked at. That is not always the case. CHECWORKS models are on a per line or per run basis. The run –

CHAIRMAN WALLIS: Could we go back to that graph that we saw? The graph was a plot of thickness versus predicted thickness. Because if you looked at amount removed versus predicted amount removed, it seems to me the comparison will be even worse.

MR. ALEKSICK: That's correct. In fact –

CHAIRMAN WALLIS: That's what you're really trying to predict is how much is removed.

MR. ALEKSICK: Yes, that is true. And my point is that in some subsets of the model, the one that we looked at here which was high pressure extraction steam, the correlation between measured and predicted is not so good. And in some subsets of the model, the correlation is much better.

CHAIRMAN WALLIS: It looks to me that in some cases it's predicting no removal whereas in fact there's a lot of removal. *So the error is percentage wise enormous?* {emphasis is added}

MR. ALEKSICK: *Yes, exactly* [emphasis added]

Advisory Committee on Reactor Safeguards Thermal Hydraulic Phenomena Subcommittee, January 26, 2003.

Accurate specification of inspection frequency is admitted by Entergy as potentially containing enormous errors. Accurate inspection frequency is the key to a valid FAC management program. Entergy proposes, through reference to NUREG 1801, to use a computer model called CHECWORKS

to determine the scope and the frequency of inspections of components that are susceptible to FAC.

Accurate specification of scope and inspection frequency is the key to a valid FAC management program. Entergy proposes, through reference to NUREG 1801, to use a computer model called CHECWORKS to determine the scope and the frequency of inspections of components that are susceptible to FAC. Entergy also provides scope of the FAC program by a by inference and directly from the LRA only to include limited piping scope.

License Renewal Application Table 3.4.1 ¶ 3.4.1-29, and Appendix B § B.1.13 (stating that management of FAC is per NUREG 1801, which in turn recommends CHECWORKS) does not meet the requirements of CFR 54.22 Because the Indian Point 2 plant recently increased its operating power level by approximately 5%, and experienced and unprecedented steam generator tube rupture event. The profiles required for CHECWORKS and the grid check points are unsubstantiated based upon these two significant changes. Changing plant parameters including coolant flow rate, the CHECWORKS model cannot be used to determine inspection frequency at Indian Point2. CHECWORKS is an empirical model that must be continuously updated with plant-specific data such as inspection results. Once "benchmarked" to a specific plant, it makes accurate predictions so long as plant parameters, such as velocity and coolant chemistry, do not change drastically. It would take as

much as 10 or more years of inspection data collection and entry to the model to benchmark CHECWORKS for use at Indian Point 2.

The Applicant has a track record of broken pipes due to corrosion, the steam generator failure a design basis accident in spite of a very low Probabilistic Risk Analysis (PRA) prediction rate. Thus, Probabilistic Risk Analysis (PRA) or pipe failures are by themselves unacceptable, and the Applicant's technical basis for a program that prevents pipe rupture or component failure as described in the LRA is inadequate to meet the requirements of 10 CFR 54.21 and other parts of 10 CFR 50.

Based on the proposed program to monitor and manage FAC, Entergy cannot assure the public that the minimum wall thickness of carbon steel piping and valve components will not be reduced by FAC to below ASME code limits during the period of extended operation.

Finally wear limits acceptance criteria are inconsistent with industry guidance and precedence regarding LRA acceptance, and SER approval for other facilities. Therefore, the NRC must deny approval of the Applicant's LRA, because it does not include an adequate plan to monitor and manage the pipe FAC as required by 10CFR54.21(a)(3) and 10CFR50.

P. Contention 20: Leak-Before-Break analysis is unreliable for welds associated with high energy line piping containing certain alloys at Indian Point 2.

Issue Statement: Stakeholders contend that the Leak-Before-Break (LBB) analysis in the Applicant's LRA is unreliable and does not provide an adequate aging management plan.

The Leak-Before-Break (LBB) concept is associated with the nuclear power plant design principles with respect to pipe failures and their safety implications. It has been introduced as a means of **partially relaxing** the requirements concerning postulated double-ended guillotine breaks. During the past few years, Leak-Before-Break (LBB) has received increasing applications as a criterion for assessing or upgrading the safety of existing plants whose provision against double-edge guillotine breaks presents deficiencies compared to current requirements.

Technically, the Leak-Before-Break (LBB) concept, defined hereafter, means that the failure mode of a cracked piping is a leaking through-wall crack which may by timely and safely detected by the available monitoring systems and which does not challenge the pipe's capability to withstand any design loading. The concept relies on experiences that double ended breaks and other catastrophic failures of primary circuit piping are extremely

unlikely. Various design, operation, inspection and monitoring aspects have been considered as prerequisites.

In recent years and months. Indian Point 2 has had a disturbing track record regarding pipe integrity issues, as evidenced by the below time line as reported in the area's paper of record, the Journal News:

> September 20, 2005: NRC and Entergy notify the public that radioactive water is leaking from IP2's spent fuel pool. The leak was discovered by contractors excavating earth from the base of the pool in preparation for the installation of a new crane, for use in transferring spent fuel from the pool to dry cask storage. NRC later admits that Entergy first discovered the leak twenty days earlier, but did not believe it was serious enough to warrant public notification. NRC orders a special inspection to determine the source of the leak.
>
> October 5, 2005: Entergy notifies the NRC that a sample from a monitoring well located in the IP2 transformer yard shows tritium contamination that is ten times the EPA drinking water limit for the radionuclide, and is consistent with tritiated water. The NRC also states in its report that the monitoring well had not been checked since its installation in 2000, following the transfer of IP's ownership from ConEd to Entergy.
>
> October 18, 2005 : The NRC and Entergy confirm that the radioactive leak discovered in August is greater than initially believed. The radioactive isotope, tritium, has been discovered in five sampling wells around Indian Point 2, while the leak at the spent fuel pool has increased to about two liters per day.
>
> November 26, 2005 : The tritium leak at IP2 remains unsolved, nearly three months after its discovery. Entergy's use of underwater cameras and divers to visually inspect and test for leaks at three locations on the steel liner's surface yield no results. Entergy must now employ different cameras to inspect the liner near the bottom of the pool, where the radiation is too high for a human diver to enter.

December 1, 2005 : IP2 Spent Fuel Pool shows tritium levels in the groundwater at thirty times the EPA limit, the highest level of tritium contamination yet discovered. In addition, the NRC announces that preliminary tests of tritiated water found in the IP1 Pool Collection System contain too much tritium to be from the IP1 Pool, suggesting that tritium-laced water is being collected in the IP1 Drain from another, unknown source. The NRC and Entergy do not know where the leak is coming from, how long it has been leaking, or the extent of groundwater contamination under the plant.

December 24, 2005 A faulty valve seal that regulates the flow of nonradioactive water to one of the plant's four steam generators causes an unplanned shutdown.

August 24, 2006 Faulty valves trigger shutdown of Indian Point 2 Drainage problem developed with discharge valves in a 10,000-gallon tank of nonradioactive water.

December 1, 2006 A 1-inch steel alloy pipe that leaked non-radiated steam and water in the containment building that houses the nuclear reactor is repaired.

March 1, 2007 Control room operators unexpectedly shut down the Indian Point 2 nuclear power plant for the fifth time in 15 months after water levels in its steam generators suddenly dropped below normal.

April 4, 2007 A steam generator problem prompted workers to manually shut down the nuclear plant.. A problem with one of the two main boiler feed pumps that send water to the plant's steam generators malfunctioned and left water levels too low.

April 7, reported on 24, 2007 A new leak of the radioactive isotope tritium was discovered at Indian Point, coming from an underground steam pipe near the Indian Point 3 turbine building.

May 14, 2007 Tritium is found in the plants sewer pipes.

May 30, 2007 Indian Point 2 interrupts power production due to steam generator problems . The broken water valve is part of a system that

feeds water to four generators, producing the steam that turns turbines to make electricity.

September 7, 2007 an alleged pinhole sized leak in a conduit is found. In fact it is a leak in a 20-24 inch fuel transfer pipe, is leaking radioactive effluent.

One prerequisite is that locations of piping systems that are susceptible to stress corrosion cracking do not qualify for Leak-Before-Break (LBB) relief. Previously, butt welds associated with 82/182 alloys for example were considered to be free of SCC problems since PWRs operate in low oxygen environments. However, more recent events with these welds have made use of Leak-Before-Break (LBB) questionable for these weld alloys. These include VC Summers, and other PWR plants.

Industry guidance as well as emerging regulatory funded studies memorialized in a NUREG "Conference on Vessel Penetration Inspection, Crack Growth and Repair" have specifically warned against traditional reliance of Leak-Before-Break (LBB) credited in Section 4.7.2 of IP2 Section 4 LRA, in spite of the nickel-based alloy weld. [page 4.7-2 of the LRA].

Indian Point 2's LRA does not respond to this potential safety threat, and relies wholly on previous studies such as WCAP-10977m and WCAP-10931. These studies are out of date. See for example, NUREG/CR-6936.

"Probabilities of Failure and Uncertainty Estimate Information for Passive Components – A Literature Review."

In addition, the NRC announced on March 13, 2007, the licensees of 40 pressurized water reactors will raise levels of vigilance concerning reactor coolant system (RCS) welds. The US Nuclear Regulatory Commission (NRC) has issued Confirmatory Action Letters (CALs) confirming the licensees' commitment to put in place "more timely inspection and [weld] flaw prevention measures, more aggressive monitoring of RCS leakage, and more conservative leak rate thresholds for a plant to shut down to investigate a possible [coolant water] leak." The measures should be put in place and welds inspected during an outage before the end of 2007. If no outage is scheduled this year, they must justify an extended schedule to the NRC.

The concerns are centered on welds containing Alloy 82 and Alloy 182, used to weld together alloys like Inconel 600 and 601 as well as dissimilar metals such as carbon steel and stainless steel. The steps were taken after the discovery of certain flaws in the welds of the pressurizer at the Wolf Creek plant, which "were repaired and did not affect the safe operation of the plant." The CALs are an interim measure while the American Society of Mechanical Engineers updates its Boiler and Pressure

Vessel Code, which will subsequently be reviewed and incorporated into NRC requirements. See Declaration Number of Ulrich Witte contained in exhibit II.

Therefore Stakeholders reiterate that the NRC must deny the Applicant's LRA for Indian Point 2 because it does not contain a reliable and adequate Aging Management Plan with regard to piping and welds, specifically Leak-before-Break (LBB), which puts at risk public health and safety during the 20 year new superseding license.

M. Contention 19 through 22. The License's ineffective Quality Assurance Program violates fundamental independence requirements of Appendix B, and its ineffectiveness furthermore triggered significant cross cutting events during the past eight months that also indicate a broken Corrective Action Program, and failure of the Design Control Program, and as a result invalidate statements crediting these programs that are relied upon in the LRA.

Issue Statement: Stakeholders assert that the Applicant's ineffective Quality Assurance Program violates fundamental independence requirements of 10CFR50 Appendix B, and its ineffectiveness furthermore triggered significant cross cutting events ***during the past eight months*** that also indicate a ***broken Corrective Action Program, and failure of the Design Control Program.***

The result of the cross cutting, inadequate programs included failures to incorporate issues such as design control breakdown, that resulted in

contaminated coolant spillage of 385-500 gallons, incorrect sections of piping cut during plant modifications, and indication of a lack of trust in employees to come forward in identifying safety culture related issues.

Specific failures included for example during the second quarter of 2007, inadequate procedures in violation of appendix B, criterion V, "instructions, procedures and drawings," during an attempt to clear interference of sumps while implementing modifications to vapor containment and recirculation pumps on march 7, 2007. The root cause is cited as "human performance error", yet multiple barriers of supervision, oversight, and flawed instructions conflicting from the work package. The root cause appears to not support the quality failure that the work package itself failed to ensure, that the piping interference was correctly planned and selected for cutting. This failure could have caused severe injuries to the work crew involved. This is an example of a cross cutting issue, were the root cause is improperly attributed, and the quality assurance failure appears to not be addressed. See inspection report 2007002. exhibit KK

A second example is Entergy's ineffective quality assurance program which should have easily caught a trend of deficient procedures associated with temporary modifications. In this example, temporary modifications were

being implemented that affected normal control lighting power. The procedure lacked general precautions, limitations, and prerequisites to prevent low lighting condition, such that operators did not have adequate lighting to monitor control panels. Yet again, the root cause was attributed to human performance, as opposed to a programmatic, symptomatic cross cutting failure. The lack of fundamental controls on the temporary modification process, lack of supervisory oversight to ensure adequate procedures with basic and generic contents to protect the health and safety of the workers, as well as the lack of safe configuration of the plant during the modification should have been caught at multiple levels, including an independent and empowered Quality Assurance Program. *Id.*

A third example is a failure to establish adequate corrective actions associated with monitoring of the service intake bay level. This failure could have prevented entry into an emergency action level, and therefore endangered the health and safety of the public during a radiological emergency. This again raises a cross cutting issue of an inadequate corrective action program as well as an ineffective quality assurance oversight program. Entergy knew of the condition and yet failed to implement corrective actions until the issue was re-identified by the NRC. *Id.*

The above examples alone indicate that license renewal based upon accurate current configuration management and control of the facility is insufficient. However, a fourth example has profound significance in creating a lack of confidence that the Applicant for license renewal is addressing the actual in situ materiel conditions of the plant, it's safe operation, and sufficient controls to ensure management of the facility as it ages beyond its design life.

In this example, a safety culture assessment result set was apparently not entered into the corrective action program. This was identified by the NRC, when Entergy failed to initiate condition reports identified during a 2006 safety culture assessment. Consequently, the adverse conditions were not evaluated and appropriate corrective actions were not identified in a timely minor. This failure by itself is sufficient to indicate that Entergy has a substantial safety culture work environment failure. Confidence by those workers that risk raising safety concerns, in spite of potential retaliation, will be immediately lost. Actual condition of the plant in terms of a baseline for managing aging is unknown, and essentially invalidates those specific programs that credit the current materiel condition of the plant in addressing Sections 3 and 4 of the License Renewal Application.

The NRC must deny the Applicant's LRA because it contains an ineffective Quality Assurance Program for an Aging Management that

violates fundamental independence requirements of 10CFR50 Appendix B, and its ineffectiveness furthermore triggered significant cross cutting events during the past eight months that also indicate a broken Corrective Action Program, and failure of the Design Control Program.

N. Contention 23: (Environmental) The Applicant's LRA does not specify, as required in 10CFR50.65 and 10CFR50.82(a)(1), an Aging Management plan to monitor and maintain all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.

Issue Statement: The Stakeholder's contend that the Applicant's LRA does not specify, as required in 10CFR50.65 and 10CFR50.82(a)(1), an Aging Management plan to monitor and maintain all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.

The condition of the Spent Fuel pool at Indian Point 2 is known to be compromised. Since at least 2005, when an independent contractor working on installing a crane in order to remove spent fuel into dry cask storage stumbled upon a underground leak at the corner of the pool, the NRC, the

Applicant and the public know that leaks exist. However, the extent , location, length and quantity of the leak remains unknown. What is known is that the Applicant failed to maintain the spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components fulfill its intended function as required by 10CFR50.65 Requirements for monitoring the effectiveness of maintenance at nuclear power plants.

The requirements of this section are applicable during all conditions of plant operation, including normal shutdown operations.

> 10 CFR54.4 (a)(1) Each holder of a license to operate a nuclear power plant under §50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in Sec. 50.82(a)(1), this section only shall apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.

In the LRA for Indian Point 2 the Applicant does not propose an Aging Management Plan that adequately addresses the compromised condition of the Spend Fuel Pool #2, or an adequate Aging Management Plan to address the intended function of the pool which is the safe containment of radioactive contamination from leaking into the environment.

The spent fuel pool's 30 year old concrete and rebar, and steel liner, are currently in a compromised condition, and cannot maintain its intended function for a period of 20 more years.

In the past year, it was accidentally discovered that ongoing, unplanned, unmonitored leaks of liquid radioactive effluents, including tritium, strontium 90 and cesium 137, are leaking from Indian Point into the groundwater and Hudson River ("Radiation Leaks"). In most cases, the duration, extent, flow paths, and/or source of the Radiation Leaks, remain unknown. To date, Radiation Leaks have been discovered throughout the Indian Point 1, 2, and 3 complex. The Radiation Leaks can neither be repaired nor remediated until sources have been found.

As of the date of this submission, upon information and belief, the Radiation Leaks result from separate, and a multitude of onsite systems, structures and components in Spent Fuel Pool 2, including, the following: (A)

Cracks in spent fuel pools; ((B) Failed or degraded fuel transfer tube sleeves; (C) Cracks and fissures.

Since September 20, 2005 the integrity for the Spent Fuel Pools have been investigated by the Applicant, however to date the Applicant has not been able to identify and locate the leaks. The following is a chronology of the spent fuel problems at Indian Point:

1. September 20, 2005: the NRC and Entergy notify the public that radioactive water is leaking from IP2's spent fuel pool. The leak was discovered by contractors excavating earth from the base of the pool in preparation for the installation of a new crane, for use in transferring spent fuel from the pool to dry cask storage. Entergy first discovered the leak twenty days earlier, but did not believe it was serious enough to warrant public notification. NRC orders a special inspection to determine the source of the leak.

2. October 5, 2005: Entergy notifies the NRC that a sample from a monitoring well located in the IP2 transformer yard shows tritium contamination that is ten times the EPA drinking water limit for the radionuclide, and is consistent with tritiated water from a spent fuel pool. The NRC broadens its special inspection to include this new information. The NRC also states in its report that the monitoring well had not been checked

since its installation in 2000, following the transfer of IP's ownership from ConEd to Entergy.

3. October 18, 2005 : The NRC and Entergy confirm that the radioactive leak discovered in August is greater than initially believed. The radioactive isotope, tritium, is discovered in five sampling wells around Indian Point 2, while the leak at the spent fuel pool has increased to about two liters per day..

4. November 26, 2005 : The tritium leak at IP2 remains unsolved, nearly three months after its discovery. Entergy's use of underwater cameras and divers to visually inspect and test for leaks at three locations on the steel liner's surface yield no results. Entergy must now employ different cameras to inspect the liner near the bottom of the pool, where the radiation is too high for a human diver to enter.

5. December 1, 2005: Entergy reports to the NRC that an initial sample from a new monitoring well five feet from the wall of the IP2 Spent Fuel Pool shows tritium levels in the groundwater at thirty times the EPA limit, the highest level of tritium contamination yet discovered. The NRC still does not know where the leak is coming from, how long it has been leaking, or the extent of groundwater contamination under the plant.

6. February 24, 2007 a cracked fuel rod is found at Indian Point 2. in the reactor's spent-fuel pool .

7. On September 7, an alleged pin hole sized leak in conduit, a pipe 20-24 inches, a fuel transfer tube and a component of the Spent Fuel Pool 2 was found to be leaking.

The Applicant's license renewal application (LRA) for IP2 LLC fails to lay out, in detail, a workable aging management plans to deal with known leaks, in Spent Fuel Pool#2. The LRA, and the UFSAR's for Indian Point 2 inadequately address the currently existing, known and unknown, environmental affects of ongoing leaks from the Spent Fuel Pools, and fails to lay out a workable aging management plan for said leaks. The only plan set forth to date, with the consent of the NRC is leave the radioactive effluent in the ground, which in time will leach into further the ground water and the Hudson River.

Due to the location of the leaks on the banks of the tidal Hudson, by allowing the radioactive contamination to remain in the ground during the 20 year new superceding license period, the radioactive effluent leaking from Spent Fuel Pool #2 and other areas of the site will continue to be leached into the Hudson River, potentially harming and making unsafe the public within six communities near the tidal area of the Hudson currently using the river for

drinking water. New York City's emergency water station is located in Croton, just a few miles down River, and the County of Rockland has just received a proposal from United Water to use the Hudson River for drinking water.

Any other business or industry, such as a dry cleaner, gas station or chemical plant, that was leaking pollution into the groundwater and river, would be immediately fined and shut down, until all the leaks had been identified, stopped and fully remediated. By even considering the Applicant's LRA for an new superseding license of 20 years, prior to a comprehensive remediation of the Radiation Leaks, the NRC has clearly surrendered its role as a regulator, and has violated it's mandate to protect public health and safety.

Neither the Applicant nor the NRC have identified an adequate aging management program for the various known and unknown leaks, thereby endangering public health and safety, by permitting unregulated radioactive waste to continue to be released into the environment during the 20 year new superceding license period. This is not only an acceptable Aging Management Program issue, but also is indicative of irresponsible and negligent management by the Applicant and improper oversight by the regulator.

Therefore Stakeholder's assert the NRC cannot approve the Applicant's LRA until the integrity of Spent Fuel Pool #2 and other components/systems are restored, and the leaks from Spent Fuel Pool #2 and other sources are fully remediated.

O. Contention 24: (Environmental) The LRA, and the UFSAR's for IP2 inadequately address the currently existing, known and unknown, environmental affects and aging degradation issues of ongoing leaks, and fails to lay out workable aging management plans for said leaks and systems imperative for Safe Shut down and cooling of the reactor.

Issue Statement: Stakeholder's assert that the Applicant's License renewal application (LRA) for IP2 LLC fails to lay out, in detail, a workable aging management plans to deal with known leaks, in the underground pipes, steam pipes and other systems critical to Safe Shut Down of the reactor, and cooling of the spent fuel pool. The LRA, and the UFSAR's for IP2 inadequately address the currently existing, known and unknown, environmental affects of ongoing leaks, and fails to lay out a workable aging management plan for leaks. Examples of inadequately addressed aging management issues which are poorly stated, vague and ambiguous include but are not limited to:

1. The reactor's coolant pump seal provides a critical leakage barrier between the pressure boundary and numerous rotating parts that seals

the pressurized reactor used in primary coolant systems. IP's LRA fails to provide adequate proof of a proper safety analysis of this critical seal, nor does it provide a detailed aging management plan, despite industry knowledge of leakage associated with this critical component. Unexpected and/or abnormal shaft movement or misalignment can introduce motions including but not limited to shaft tilt, radial offset and orbit, and depending on the magnitude and scope of this displacement, and thus the seal arrangement, creates potentially dangerous site specific operational issues of concern, and site specific wear (aging) effects that must be accounted for with a detailed site specific aging management plan.

2. It appears from IP2's LRA that applicant contends the feedwater heater is outside the scope of License Renewal. We disagree. The feedwater heater is a crucial component in maintaining thermal performance, but more importantly, aging issues unchecked contribute greatly to INCREASED pipe fatigue and failure, which in turn increases leakage issues for key component pipes in the reactor system. Simply stated, loss of feedwater will impose SEVERE STRESS on the entire plant in terms of increased heat flux in the fuel, and greatly increased (and associated fatigue) on feedwater nozzles, headers, and piping. 41. U.S. Nuclear Regulatory Commission, "Rates of

Initiating Events at U.S. Nuclear Power Plants 1987-1995", NUREG/CR-5750, February 1999.

3. Various piping industry sources place the life expectancy of stainless steel pipes as little as 20 years without proper chemistry controls, and cumulative usage factors being improperly analyzed under finite element analysis and other mechanistic based failures often due to improper maintenance of the system. IP2 is now in its 33rd year of licensing. There exists no detailed aging and maintenance plan provides indication of adequate management of chemistry, or fundamental maintained requirements such as those required in 10cfr50.65 in the LRA. In addition, there are no commitments that provide a viable and workable pipe or component replacement strategy for key component pipes needed for the cooling and safe shut down of the reactor.

Unplanned, unmonitored leaks of liquid radioactive effluents, including tritium, strontium 90 and cesium 137, are leaking from Indian Point into the groundwater and Hudson River ("Radiation Leaks"). In most cases, the duration, extent, flow paths, and/or source of the Radiation Leaks, remain unknown. To date, Radiation Leaks have been discovered throughout the Indian Point 1, 2, and 3 complex. The Radiation Leaks manifestly can neither be repaired nor remediated until sources have been identified and/or located.

As of the date of this submission, upon information and belief, the Radiation Leaks result from separate, and a multitude of onsite systems, structures and components, including, the following: (A) Failed or degraded pipes (including pipes that transport liquids and pipes which transport steam); (B) Cracks in spent fuel pools; (C) Failed or degraded valves; (D) Reactor vessel failed welds in the bottom or vessel (which inspectors have been unable to adequately view and reach); (E) Pinhole leaks around weld joints; (F) Failed or degraded gauges; (G) Failed or degraded fuel transfer tube sleeves; (H) Failed or degraded steam generator tubes; (I) Inadequate or improperly operating drain systems; (J) Cracks and fissures.

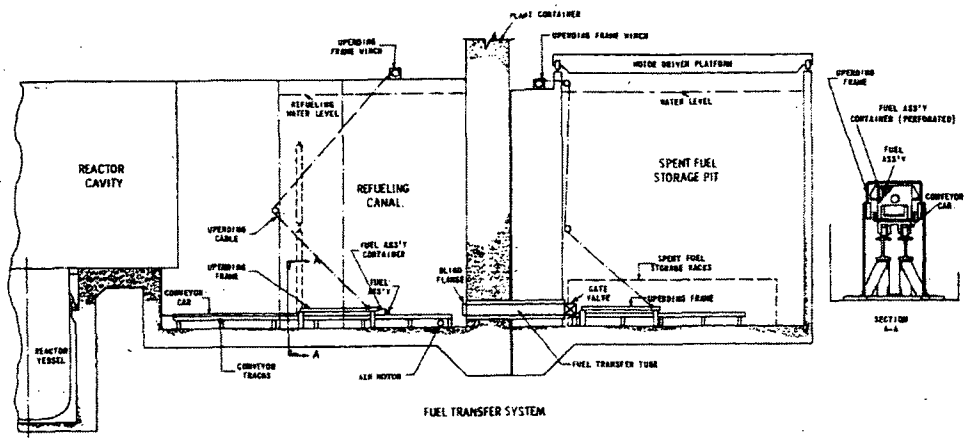
The facts provide that pipes both stainless and carbon alloy are cracking and breaking at Indian Point 2. For example, only recently on September of 2007, Entergy admitted to finding a leak in the conduit that is a part of the fuel transfer canal between the reactor and the spent fuel pool. The article in the Journal News stated in part:

> By BRIAN J. HOWARD
> THE JOURNAL NEWS
> (Original publication: September 7, 2007)
>
> BUCHANAN - Workers have discovered a pinhole-sized leak in a ***conduit*** used to transfer spent fuel from the reactor to the containment pool at Indian Point 2.

The leak was found Wednesday during testing for groundwater contamination from leaks of radioactive tritium and strontium 90 that were first discovered in 2005.

"It appears that there is a potential pinhole leak in the fuel transfer canal, which we believe could be a contributing source to the groundwater contamination that we've been talking about," said Jim Steets, a spokesman for Entergy Nuclear Northeast, the plant's owner.

A conduit is commonly understood to be an electrical conduit on order of 1or 2 inches in diameter. However this is a gross misrepresentation, as the pipe in question is in fact a 20-24in pipe. There is a world of difference between conduit and pipe, and the Applicant and NRC clearly know the difference. Entergy's representative purposely released misleading information to the public, when he alleged it to be pin hole sized leak in Further, the leak is more than likely, to be a leak in the FUEL TRANSFER TUBE, which may have a much greater impact on the integrity of the facility.



CONSOLIDATED EDISON INDIAN POINT UNIT No. 2	
UFSAR FIGURE 9.5-1	
FUEL TRANSFER SYSTEM	
MIC. No. 1999MC3886	REV. No. 16A

It is worthy of note that irradiated water from this recently discovered leak, and all the other leaks flow into fissures in the bedrock under the plant, and will eventually leach into the tidal Hudson River. Many of the cracks and fissures in the bedrock were created when the bedrock was blasted as the plant was first built, and therefore the irradiated effluent can take a very convoluted route into the environment, the groundwater and the Hudson River.

The multiple leaks at Indian Point 2 provide direct evidence of underground pipe failure and/or degradation due to the aging of various systems. Such systems are not being adequately inspected or addressed by the applicant, and proof that the applicant's management of aging issues is wholly inadequate.

In fact, certain Radiation Leaks, including tritium leaks allegedly from underground pipes on the "non-radioactive" side of plant were discovered purely by random accident on April 7, 2007, rather than via a coordinated, intelligent aging management and inspection plan. Other leaks were discovered, only because special excavation work being done by a contractor led to investigations after tritium contaminated water was found seeping from surface cracks in spent fuel pool number 2, not through regular inspection and maintenance . In fact the length of time and extent of the Radiation Leaks have existed remains unknown.

The multiple leaks are symptomatic of an aging system, that was not properly and comprehensively inspected and maintained during the initial license period, There is no reason to believe that during the 20 years of the new superceding license the Applicant will do a better job of properly inspecting, maintaining and managing the aging facility. Nor does the LRA identify an aging management plan to locate, stop and remediate the current and future leaks. There are only vague reference to best industry standards, and sparsely defined sketches of potential aging management plans to deal with leakage issues caused from corrosion, fatigue, thermal shock, FAC (flow-accelerated corrosion), and other leakage causes of concern during the 20 year period of license renewal.

At the Kashiwazaki plant in Japan, in July 2007, radiation leaked into the environment through a small hole, then flowed along electrical cabling, then into an air conditioning duct, then into a drainage ditch, and then finally out into the sea.) The existence of the Radiation Leaks provides direct evidence of underground pipe failure and/or degradation that has not been adequately addressed by the licensee. Ordinary maintenance failed to reveal the specific locations of numerous Radiation Leaks, therefore the limited aging management programs indicated in the LRA will also fail to identify radiation leaks before they cause damage to the environment, or before the

leaks become breaks. As example, there is no aging management plan to address known potential pipe bursts in piping adjacent to plugged tubes in IP2's LRA. Further the LRA does not specify comprehensive visual inspections, vacuum testing and ultrasonic testing for all pipes, including buried pipes to determine corrosion, failure, environmental fatigue and other aging affects.

Moreover, at an April 26, 2007 public NRC meeting in Cortlandt, N.Y. ("April NRC meeting"), NRC and Applicant representatives conceded that they did not even know the metallurgic composition of much of the underground piping. Without a complete and comprehensive knowledge of the composition and layout of the underground piping system the Applicant will be unable to implement an adequate aging management plan. Inaccessibility limits the inspection and testing of substantial segments of these aged and leaking pipes and components which play crucial roles in the cooling and safe shut down of the IP2 reactor, especially those having a buried or embedded environment. Thus, the Applicant cannot assure the NRC and the public that they will be able to manage effects of aging, soil elements, the intake of brackish water from the Hudson River and/or storm surges during the 20 year new superceding license period, which have already caused dangerous corrosion of Indian Point's entire piping, valve and gauge system

resulting in the current leaks. It is further noted, that IP2 has not addressed the unique corrosion issues associated with the use of brackish water in the coolant process.

In the past few years there has been a significant increase in the amount of leaks found, at IP2, which indicates that as the plant ages there will be increased frequency of pipe leakage during the 20 year period of license renewal. Since August 2005 the Applicant has not been able to identify the source of the leaks, the duration of the leaks. On December 1, 2005, the applicant reported to the NRC that an initial sample from a new monitoring well five feet from the wall of the IP2 Spent Fuel Pool shows tritium levels in the groundwater at thirty times the EPA limit, the highest level of tritium contamination yet discovered. In addition, the NRC announces that preliminary tests of tritiated water found in the IP1 Pool Collection System contain too much tritium to be only from the IP1 Pool, suggesting that tritium-laced water is being collected in the IP1 Drain from another, unknown source. The Applicant still does not know where the leak is coming from, how long it has been leaking, or the extent of groundwater contamination under the plant.

August 24, 2006 Faulty valves trigger shutdown of Indian Point 2

Drainage problem. Workers shut down Indian Point 2 yesterday morning after problems developed with discharge valves in a 10,000-gallon tank of nonradioactive water.

November 29, 2006: An unplanned shutdown at Indian Point 2, because a 1" steel alloy pipe was found leaking non-radiated water in the containment building.

April 24, 2007: A new leak of the radioactive isotope tritium has accidently discovered at Indian Point, coming from an underground steam pipe near the Indian Point 3 turbine building, company officials and federal regulators confirmed yesterday.

September 7, 2007 a pinhole leak As recently as September 7, 2007a pinhole leak in the fuel transfer canal, was found which may be a contributing source to the ongoing groundwater contamination of Strontium and Tritium.

All of the Radiation Leaks point to the imperative necessity for a complete inspection and comprehensive corrosion analysis of all underground and critical in scope piping systems and associated equipment that contribute to significant aging, fatigue, corrosion and vibrational degradation . Compromised pipes can cause or fail to mitigate a serious accident, including a core damage event. Therefore, to properly maintain the aging facility any and all compromised pipes must be replaced, including but not limited to, the

ones under the reactor where information from discussions with Indian Point workers leads us to believe seals may be leaking.

The insufficiency of a reliable aging management program in the LRA of IP2 LLC increases the exposure risks of plant workers during the 20 year period of license renewal, and greatly increases the potential for a significant nuclear incident at the Indian Point facility during the period of license renewal, as increasing leak rates negatively impinge upon the core cooling component structures, and increase the risk of severe pipe ruptures that would lead to a release of unmonitored and uncontrolled radioactive contaminants into the environment, including the Hudson River, thus presenting a significant and increased risk to public health and safety.

The NRC itself has expressed concerns on this very issue as relates to ALL license renewal requests, and requested as a part of the license renewal application process that their licensees perform an assessment to ascertain and/or determine the potential severity of the effects of reactor water coolant environment on fatigue. Further, where appropriate, the NRC further suggested license renewal applicants provide a proper aging management plan to deal with said fatigue issue. This concern was/is included in discussions found in NUREG/CR-6674.

The Applicants in their LRA for IP2 LLC make a brief reference to reliance on a nuclear regulator approach to this significant issue, yet fail to identify with specificity an aging management plan which deals with the unique site specific environmental effects at the Indian Point facilities. The adequacy, or lack there of, as relates to this specific aging management issue is a matter of fact, that can only be resolved after interested parties, including community Stakeholders have an opportunity to submit evidence, cross examine expert witnesses, and a conduct a full review of Entergy'supporting and/or discovered documents and a full in depth review has been conducted on the part of the hearing board.

Entergy's Indian Point facility (IP1, IP2 and IP3) have numerous serious leak issues. It is further known that leaks in the cooling pipes (critical components in the reactor water coolant process) present a serious plant specific safety issue/problem if an adequate aging management plan is not in place. Currently it is not. Maintenance logs and other documents that will be found in pre-hearing document discovery will prove IP2 and IP3's aging management plan for this issue is woefully inadequate. Further, there are numerous NRC inspection documents identifying leak issues at the plant which will support this contention. The NRC and the nuclear industry have admitted that environmental fatigue will increase the rate, volume and number

of these leaks during the period of 20 years of additional operation of these aged facilities

The industry's newly developed and unproven approach to this known aging issue is inadequate, and fails to adequately address the unique environmental issues specific to IP2 , as said plants rely upon a unique brackish water supply for their reactor core cooling system.

Generic industry approach is inadequate to address the unique site specific leaks in the pipes, as evidenced by various already identified leaks. Leaks are a precursor to PIPE BURSTING in nuclear reactors primary coolant systems. See Declaration of Ulrich Witte, exhbit yy

IP2's poorly defined and inadequate aging management issues as relates to this specific issue greatly increases the chances of a significant incident. such as large pipe burst, that could lead to an off site release of radioactive contaminants, thus creating a significant risk to human health and the environment, if as is contended here, said aging management plan is inadequate to properly address this aging management issue.

The NRC and Entergy do not have an aging management plan for the underground Radiation Leaks, thereby endangering the public's health and safety, by permitting unregulated radioactive waste to continue to be released into the environment during the 20 year new superceding license period. Not

only is lack of an adequate aging management program at issue, but also it is indicative of irresponsible and negligent management by the Applicant, Entergy, and improper oversight by the regulator, the NRC.

The Applicant initiated actions to pump out the Unit-1 Containment Spray Sump through a filter/demineralizer system, designed to remove Sr-90, and investigate the source and means of the Sr-90 groundwater contamination. This fact raises the question, is Entergy in violation of the terms of their SafeStor for IP1. When the applicant started to remove the underground leaks by pumping the radioactive contamination out of the ground, it caused more radioactive material to be released.

Therefore the NRC ordered that the Applicant to stop removing the radioactive effluent from ground, and to only monitor . See exhibit JJ Due to the location of the leaks on the banks of the tidal Hudson, by allowing the radioactive contamination to remain in the ground during the 20 year new superceding license period, the radioactive effluent will continue to be leached into the Hudson River, potentially causing great harm to human life, as 6 communities within the tidal area of the Hudson currently use the river for drinking water, New York City's emergency water station is location in Croton, just a few miles down River, and the County of Rockland has just

received a proposal from United Water to use the Hudson River for drinking water.

Critically, compromised pipes can cause or fail to mitigate a serious accident, including a core damage event. Therefore effects of or associated with aging – including embrittlement, corrosion, rust, heat, and microbiological and chemical agents – may destabilize and weaken the tensile strength of the piping and associated equipment and components. This presents an unacceptable risk during an extended life of the plant which must be specifically and fully addressed by the aging management program. The aging management plan iterated in the Indian Point application utterly fails. .

The Applicant has displayed plume maps of the strontium 90, tritium and cesium which is pooling underground due to the ongoing leaks, but have claimed the maps to be proprietary, in addition a few weeks after the deadline for Intervener Petition's the applicant will deliver a new leak report. Therefore, once again, FUSE respectfully requests the opportunity to amend this contention or summit new contentions after the new leak report and plume maps are made available to the public Stakeholders be granted.

FUSE contends that the NRC must deny the Applicant's LRA because it fails to adequately address the current Radiation Leaks, and fails to provide

an effective and adequate Aging Management Plan with regard to future Radiation Leaks, and therefore adequately protect public health and safety, and the environment.

Supporting Document References for This Contention

1. NUREG/CR-5999 (ANL-93/3), "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," April 1993.

2. NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

3. NUREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998.

4. NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

5. NUREG/CR-6674 (PNNL-13227), "Fatigue Analysis of Components for 60-Year Plant Life," June 2000.

6. U. S. Nuclear Regulatory Commission, Generic Safety Issue 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

P. Contention 25: The Applicant has failed in its LRA to include as part of the EIS Supplemental Site Specific Report any refurbishment plans in order to meet the mandates of NEPA, of NRC 10CFR 51.53 post construction environmental reports or of NRC 10CFR 51.21.

Issue statement: Stakeholders assert that the Applicant's LRA fails to comply with 10CFR 51.21 and 10CFR51. 23, by failing to provide

refurbishment plan, for already planned refurbishment during the proposed 20 year new superseding license.

The Applicant is required in its EIS Supplemental Site Specific Report required to fulfill the requirements of NEPA, and codified in 10 CFR Rules and Regulations as defined in 51.21 and 51.53 requires NRC licensees filing a LRA for the purpose of license extension to include as a part of the EIS Supplemental Site Specific Report any refurbishment issues/plans and the environmental risks associated with said refurbishment. The Applicant by evidence provided below failed to comply with this rule.

In the Applicant's filed LRA for Indian Point 2, in Appendix E, Supplemental Environmental Report, section 3.3 of it's Environmental Report Refurbishment Activities, the Applicant simply and dismissively states that 'there are no such refurbishment activities planned and/or anticipated at this time' and thus provide the Nuclear Regulatory Commission no Environmental Report on refurbishment. By claiming that there are no refurbishment activities planned, the Applicant indicates that there are no environmental concerns which need to be addressed in the LRA.

However, the Applicant omitted the fact that it had already prepared for a major refurbishment by ordering a Replacement Reactor Vessel Heads for Indian Point #2, with delivery date scheduled for October 2011, as evidenced

by the attached page (a true and accurate copy of the PDF web based file) of the Doosan Heavy Industries Construction Co., Ltd presentation at the Burns & Roe 17th Annual Seminar, Powering the Future, March 21, 2007 and contacted the engineering and construction required for this substantial refurbishment. Attached hereto as Exhibit LL and rewritten below:

Entergy Replacement Reactor Vessel Head

(A) Customer: Entergy

(B) Projects: ANO #2 (Site Delivery: January, 2008), Waterford #3 (Site Delivery: February, 2008), Indian Point #2 (Site Delivery: October, 2011), and Indian Point #3 (Site Delivery: October, 2012)

(C) Primary Contractor: Westinghouse

(D) Scope: Four (4) RRVHs

(E) Two (2) sets of CRDM (for Indian Point #2 & 3 only)

(F) Manufacturer: DOOSAN (EMD supplies CRDM as the sub supplier)

The plans to potentially replace the reactor head for Indian Point Unit 2 and 3 as well as the CRDMs is costly—of order of 15-20 million dollars per unit. The applicant only purchased these heads for Indian Point and two other facilities. Not for the entire fleet. FUSE asserts that these plans even if actual

installation date is not established, or even if the modification is potentially firm at this point that the Stakeholders are entitled to more than just mere silence on this issue.

The Doosan presentation is clear evidence of the Applicant's plans for refurbishment. Refurbishment on the scale of a reactor head replacement, which has already been ordered and with a specific delivery date makes this omission by the Applicant deliberate. Hundreds of people are involved in a decision to replace a reactor vessel head, and requires senior management approval of such a costly refurbishment. Since at least 2003, boric Acid corrosion and rust in the reactor vessel head were degradation issues known by the Applicant, and may be major contributory factor in the Applicant's decision to plan the significant refurbishment of reactor vessel head replacement

The Applicant is a multinational corporation with extensive knowledge and expertise in the nuclear reactor industry and with ownership rights to eleven nuclear reactors in America. Therefore the omission of this significant and already planned reactor refurbishment during the proposed 20 year new superseding license, from the Supplemental Environmental Report attached to the Applicant's LRA as Appendix E was neither accidental, nor a mere oversight in compilation of its License Renewal Application.

Further, the Applicant offers itself up as a supplier of expert assistance in the filing of LRA's to other NRC licensees considering a 20 year license renewal for their own facilities.

1. Basis for Contention

(i) Therefore, Stakeholder's content that the Applicant, the second largest reactor owner in the United States, deliberately hid material facts, and egregiously submitted a materially false LRA, in a violation og 10 CFR50.5 and 10CFR50.9,by attempting to hide significant environmental, health and safety concerns in an attempt to streamline approval of it's LRA, that could greatly impact the safety of the Stakeholder's community.

(ii) The Applicant has not fulfill its legal obligation as delineated in NEPA reference and the Code of Federal Regulations reference to prepare and submit, as part of their applications, a description of the proposed refurbishment actions, including any plans by the Applicant 'to modify the facility' and describe in detail the modifications affecting the environment or affecting plant effluence that affect the environment' 10CFR 53(c) (1)(2).

(iii) Moreover, 10CFR 5 (c)(3)(ii)(E) mandates that 'all license renewal applicants shall access the impact of refurbishment and other license renewal related construction activities on important plant and animal habitats. Additionally, the Applicant shall assess the impact of the proposed action on

threatened or endangered species in accordance with the Endangered Species Act'.

(iv) Replacement of a reactor vessel head for Indian Point 2 is not only a refurbishment issue, but a significant environmental issue that affect public health and safety on many levels, and that must be evaluated during the license renewal process. The means and method of disposal of the irradiated old reactor vessel heads must be addressed, in the Aging Management Plan. Indian Point was not designed, nor licensed to act as a radioactive waste storage facility, however with the closing of Barnwell to Indian Point radioactive waste streams beginning in 2008, the impacts of any and all radioactive waste streams, including disposal of old reactor vessel heads, generated at Indian Point, are an issue of paramount importance for the safety the Stakeholder community.

(v) The Applicants have failed to provide the mandated reports specificity required, and have also failed to provide environmental reports required with regard to its plans to change or modify the facility or refurbish same.

(vi) As Stakeholders living within 3, 10 or 50 miles of the Indian Point facility owned by the Applicant any reactor refurbishment issue that

contributes to any potential environment, health or safety risks is of great concern.

Hiding or ignoring significant information is in contradiction to the NRC regulations which requires LRAs to be complete, accurate and truthful. The NRC must revoke it's acceptance of the Applicant's LRAs as complete and accurate, and further take administration legal action to hold the Applicant accountable.

2. Contention is Within Scope in the License Renewal Process

The reactor core coolant system, and all its primary parts, including piping are within the scope of the license renewal process, as is the reactor vessel head. By proxy, and by NRC regulation, planned refurbishment of the reactor head for Indian Point 2 is within scope. Therefore, this contention brought by the Stakeholders against Indian Point 2 regarding refurbishment is within the scope of Entergy's License Renewal Application.

3. Contention Raises Material Issues of Fact and/or Law

There exists issues of fact and/or law in this contention. The reactor vessel head replacement is never a like-for-like switch of components or equipment, and is one of the most critical refurbishments that a reactor licensee can undertake, In some situations replacement of the reactor vessel head may require cutting a hole into the containment.

1. Reactor vessels are far beyond tangential components. They contain the nuclear fuels in the plants, and, over time, are irradiated which can lead to embrittlement, deterioration, loss of material, and less able to withstand flaws which may be present.

The 2002 incident at the Davis Besse Nuclear Plant highlights the integral nature of the vessel and the vessel heads. Despite this vast knowledge pool, the Applicant neglected to list, describe or report the vessel head replacement, or any other refurbishment actions in the environmental supplement of the LRA and marked as Appendix E.

2. The omission of significant refurbishment issues from the EIS Appendix E cause Stakeholders to claim that the Applicant has egregiously taken the position that the above changes and reactor modifications are not within the purview of the LRA application, in violation of NRC regulations. The refurbishment of the vessel head, and other proposed changes and refurbishments necessary for the replacement of the reactor vessel head, yet undisclosed, are within the scope of 10CFR 53 and 10CFR 54.21. As stated by the NRC:

> For the purposes of the Environmental Impact Review, refurbishment describes an activity or change in a facility that is needed to support operations during the renewal term.

The replacement of the reactor vessel heads are needed to support operations during the applied for new superseding term of an additional 20 years. Further the 10CFR 53 and 10CFR 54.21 require the Applicant to include such reactor vessel head replacement in the environmental report, delineating with specificity all potential impacts, remediation, and alternatives, including but not limited to, worker radiation exposure, construction traffic and noise, construction runoff, radiation releases, impacts on plant and animal habitats, and the impact of the proposed actions on threatened or endangered species in accordance with the Endangered Species Act.

NRC places great importance on integrity and honesty in the submission of documents to the agency, to ensure trustworthiness and integrity are beyond reproach. The NRC writes,

> It is paramount to the mission of the NRC for the licensee to maintain information and communicate with the NRC in such a manner that all information is complete and accurate in all material respects to allow the NRC to complete their mission.
>
> It is the responsibility of the licensee personnel to work together to ensure the health and safety of the public and plant personnel.
>
> Effective, complete and accurate communication is required to ensure this vital goal, regardless of the potential financial or business impact.

Reactor vessel head replacement is a complex reactor refurbishment project that involves almost every major department, and 100's of personnel, including Senior Members of Management. Omission of such a significant project from the LRA applications of IP2 is a serious violation of 10 CFR 50.5 and 50.9.

(i) § 50.5 Deliberate misconduct

(A) Any licensee, applicant for a license, employee of a licensee or applicant; or any contractor (including a supplier or consultant), subcontractor, employee of a contractor or subcontractor of any licensee or applicant for a license, who knowingly provides to any licensee, applicant, contractor, or subcontractor, any components, equipment, materials, or other goods or services that relate to a licensee's or applicant's activities in this part, may not:

> (1) Engage in deliberate misconduct that causes or would have caused, if not detected, a licensee or applicant to be in violation of any rule, regulation, or order; or any term, condition, or limitation of any license issued by the Commission; or
>
> (2) Deliberately submit to the NRC, a licensee, an applicant, or a licensee's or applicant's contractor or subcontractor, information that the person submitting the information knows to be incomplete or inaccurate in some respect material to the NRC.

(B) A person who violates paragraph (a)(1) or (a)(2) of this section may be subject to enforcement action in accordance with the procedures in 10 CFR part 2, subpart

(C) For the purposes of paragraph (a)(1) of this section, deliberate misconduct by a person means an intentional act or omission that the person knows:

> (1) Would cause a licensee or applicant to be in violation of any rule, regulation, or order; or any term, condition, or limitation, of any license issued by the Commission; or
> (2) Constitutes a violation of a requirement, procedure, instruction, contract, purchase order, or policy of a licensee, applicant, contractor, or subcontractor.

(ii) 50.9 Completeness and accuracy of information.

(A) Information provided to the Commission by an applicant for a license or by a licensee or information required by statute or by the Commission's regulations, orders, or license conditions to be maintained by the applicant or the licensee *shall be complete and accurate in all material respects.*

(B) Each applicant or licensee shall notify the Commission of information identified by the applicant or licensee as having for the regulated activity a significant implication for public health and safety or common defense and security. An applicant or licensee violates this paragraph only if the applicant or licensee fails to notify the Commission of information that the

applicant or licensee has identified as having a significant implication for public health and safety or common defense and security. Notification shall be provided to the Administrator of the appropriate Regional Office within two working days of identifying the information. This requirement is not applicable to information which is already required to be provided to the Commission by other reporting or updating requirements.

Realizing the importance of public trust, and how easily it can be lost, the NRC places great importance on the completeness and accuracy in all materials submitted to them, and this standard takes on far more importance in and issue as License Renewal of a reactor, which has such large term potential impacts on a community, public health and safety.

Contention is Supported By Facts and/or Expert Opinion

The Stakeholder have met the minimal requirements of the 10 CFR rules and regulations in presenting this contention in a concise statement of the facts adequate to establish that said contention is entitled to a further and complete review of the issues contained herein. It is pointed out that the rules governing the license renewal process, and hearings lay out some basic criteria that a Stakeholder must meet to have a contention accepted for further review. *Section 2.309(f)(v) requires,*

> ...a concise statement of the alleged facts or expert opinion which support the petitioner's position on the issue and on which the petitioner intends to rely at hearing, together with references to the specific sources and documents on which the H petitioner intends to rely to support its position on the issue.

Misrepresentation in Licensee communication and documents are very serious violations of NRC Rules and Regulations. Further, the very principals of NRC's enforcement policy make it abundantly clear that significant violations of the 10 CFR rules and regulations can be subject to license suspension and/or termination.

(iii) NRC Enforcement Policy Excerpts

The primary purpose of the NRC's Enforcement Policy is to support the NRC's overall safety mission in protecting the public health and safety and the environment. Consistent with that purpose, the policy endeavors to:

(A) Deter noncompliance by emphasizing the importance of compliance with NRC requirements,

(B) Encourage prompt identification and prompt, comprehensive correction of violations of NRC requirements.

Therefore, licensees, contractors, and their employees who do not achieve the high standard of compliance which the NRC expectations may be

subject to enforcement sanctions. Each enforcement action is dependent on the circumstances of the case. However, in no case will licensees who cannot achieve and maintain adequate levels of safety be permitted to continue to conduct licensed activities.

Herein, the Stakeholder are raising very troubling issues of both fact and law. The Applicant, at best, has made a critical error which should cause the NRC to dismiss the LRA.. At worst, the Applicant has purposely attempted to omit facts, thereby misrepresenting its plan ot the NRC and the public, during the proposed 20 year new superseding license. The undersigned therefore respectfully request that the Applicant's LRA be denied due to the fatal errors in the same.

Q. Contention 26: Environmental Effects and Cascading Consequences on the Aging structures, deteriorated conditions and compromised systems, of a Terrorist Attack On Aging Indian Point Nuclear Reactors Contention are not considered in the LRA for IP2.

This Contention is written in honor of the brave men and women who gave their lives in the World Trade Center, American Airlines Flight 11, American Airlines Flight 77, United Airlines Flight 175, United Airlines Flight 93 and the Pentagon.

Stakeholders claim that the environmental effects and cascading consequences on the aging structures, deteriorated conditions and compromised systems, of a terrorist attack on Indian Point Nuclear Plant are not considered in the LRA for IP2.

On September 11th, 2001 America experienced the darkest day in our nation's history when two planes filled with terrorists flew into the World Trade Center in New York, New York.

2996 brave souls woke up to a bright beautiful sunny fall day, not knowing that in a few scant hours they would become the faces etched into our souls, the victims never forgotten, the heroes remembered and honored each and every year as America remembers our darkest hour. The lives of every American were changed that day, the destiny and direction of our nation changed forever. We were attacked on our home soil, the sacred lands of America invaded by radical terrorist bent on forcing their evil will upon a free people, using fear, intimidation and despicable terrorists attacks to bring America to its knees.

One of the hijacked planes used the Hudson River as a guide, flew directly past the twin domes of the Indian Point Reactors. Notably, the 9/11 Commission learned that the original plan for a terrorist spectacular was for a larger strike, using more planes, and including an attack on nuclear power

plants. In an Al-Jazeera broadcast in 2002, one of the planners of 9/11 said that a nuclear plant was the initial target considered.

We also know from the 9/11 Commission's investigation that, even after the plot was scaled down, when Mohammed Atta was conducting his surveillance flights he spotted a nuclear power plant (unidentified by name, but obviously the Indian Point nuclear power plant) and came close to redirecting the strike. National Research Council analyses and post-9/11 intelligence has also indicated that the U.S. nuclear infrastructure is viewed as an alluring target for a future terrorist spectacular. As the Chairman of the National Intelligence Council stated in 2004, nuclear power plants "are high on Al Qaeda's targeting list," adding that the methods of Al Qaeda and other terrorist group may be "evolving."(*Council on Intelligent Energy & Conservation Policy (CIECP) comments to proposed rule 10 CFR Parts 50,72 nd 73, regarding power reactor security requirements at Licensed Nucelar Facilities,* March 27, 2007 Re*: Proposed Rule: Power Reactor Security Requirements* (RIN 3150-AG63)

The nuclear industry, NEI and the NRC use a statistical analysis to justify eliminating the environmental effects of a terrorist attack from review and consideration in Entergy's License Renewal Applications for IP2 and IP3. Despite the ruling in Diablo Canyon's "Mother's For Peace" case the Ninth

Circuit Court ordered that the effects of a terrorist attack are to be included in the Environmental Review required by NRC regulation 10CFR 51.53 to fulfill the NRC's NEPA requirements. However, the NRC has decided to allow industry financial concerns to over ride the Agency's singular and most important goal, the protection of human health and the environment.

Even though since 9/11 an entire cabinet level department has been created and billions of tax payer dollars are being spent on Homeland Security to protect against terrorism.

The problem is, statistics, risk modeling analysis worked out on some computer do not reflect the reality that is life. As those towers came down, as New Yorkers and citizens from around the world lost their lives in the blink of and eye, NRC's assurances that and attack on a nuclear reactor were so remote as to almost not exist rings falsely in our ears.

We, the citizens of New York know better than any one that terrorists can plan, mount and carry out a successful attack on a target within the borders of the United States of America, we learned first hand how horrendous the aftermath of such an attack can be. We do not accept NRC's false assurances that a pathetic DBT, and a poorly trained private security force can keep us safe. The costs associated with the aftermath of 9/11 are far

to high to count, the loss of human life far to priceless to put a dollar value on. We can replace the energy Indian Point produces, but not the lives.

So, in honor of those fallen heroes, we the citizens of the Hudson River Valley living within 50 miles of Indian Point raise our voices as one in demanding that the environmental costs associated with a terrorist attack be included in Entergy's License Renewal Applications process for Indian Point Reactors Two and Three as was ordered by the Ninth Circuit Court of Appeals in the Diablo Canyon "Mothers For Peace" ruling.

Basis for Contention

1. As stakeholders, petitioners, and property owners living within 3, 10 and 50 miles of the Indian Point facility owned by two unique and separately owned Entergy Limited Liability Corporations (IP2 LLC and IP3 LLC) we are extremely concerned about the potential effects of any incident at the Indian Point Energy Center Site that could result in off site release of radioactive contaminants.
2. The National Environmental Policy Act (NEPA) requires the NRC to require and environmental study of the effects of given events in evaluating a licensing request on the part of their licensees. The preamble of this act reads in part:

> *"To declare a national policy which will encourage productive and enjoyable harmony between man and his environment; to*

promote efforts which will prevent or eliminate damage to the environment and biosphere and stimulate the health and welfare of man; to enrich the understanding of the ecological systems and natural resources important to the Nation..."

The law applies specifically to federal agencies and the programs they fund and/or regulate. Essentially it requires that, prior to taking any "major" or "significant" action, the agency must consider the environmental impacts of that action.

3. Entergy's License Renewal Application (LRA) for IP2 and the 20 year period of additional reactor operation it represents is a "major" or "significant" event/action on the part of a Federal Agency, therefore the rules of law and procedure found in NEPA apply to this relicensing process. NRC as and agency has accepted the reality that NEPA applies to many of the actions they take as and agency as is witnessed by their own regulation 10CFR 51.53 which was created as the NRC's implementing criteria for their agency's responsibilities in abiding by the laws and constraints found in NEPA.

4. The action forcing provision of the NEPA law requires an Environmental Impact Statement (EIS) to be written, which outlines the risks, and the costs to human health and the environment, should that risk become a reality for all major federal actions which may have a

significant impact on the environment. Further, the requirements of NEPA state that the agency (in this case, NRC) must involve the public by giving them notice and allowing them to comment on the proposal. The only exception is if the proposal falls within a previously-established "Categorical Exclusion" which is a category of actions that generally are not likely to have significant impacts. In such rare cases neither an EA nor an EIS needs to be prepared so long as the proposed action does not have any unusual characteristics that create potential for risk significant impacts.

Even if the relicensing of IP2 fell into this "Categorical Exclusion", it would still require an EIS by virtual of the unusual characteristics of nuclear reactors that raise the potential for risk significant impacts.

5. The NRC in numerous licensing activities involving nuclear facilities, specifically in relicensing actions, has wrongfully attempted to narrow the scope of the EIS. Specifically, the NRC has attempted to remove from inclusion in the EIS some crucial risks and the costs of any aftermath of such events.

A) The aftermath and significant impacts on the environment should a successful terrorist attack occur at the Indian Point Energy Facility located in Buchanan, New York. NRC wants to rely upon best estimate modeling by the self vested nuclear industry to claim the likelihood of a terrorist attack is, all but, impossible.

As citizens living in New York, the hallow land at Ground Zero acts as a constant reminder that terrorists can and will attack at any given time, and can plan, mount, launch and successfully carry out a successful attack on US infrastructure targets. The NRC cannot refute the very real fact that a large commercial aircraft commandeered by terrorists flew right past the twin domes of Indian Point on September 11th, 2001 on its journey to crash into the Twin Towers in Manhattan.

B) The aftermath and significant impacts on the environment should the Emergency Evacuation Plan for Indian Point fail to function as envisioned in the case of a significant incident or attack involving off site release of radioactive contaminants occur should also be a part of the EIS for IP2's LRA. The fact that the Emergency plan is a living fluid

> document is NOT THE ISSUE, the issue is what happens, what are the environmental costs if the plan does not work, or function as envisioned, as was/is the case in the aftermath of Hurricane Katrina. See for example the Witt Report . We are not saying the Emergency Plan itself is in scope, but the aftermath of its failure and/or non workability are within scope of this process under the rules and guidance of NEPA.

The aftermath should the NRC's DBT, which dictates the security requirements and types of events that Indian Point must be capable of defending against in the case of a security breach of any type, including but not limited to A) a significant nuclear incident leading to a major release of radioactive contaminants, B) a terrorist attack, or C) a successful action by malcontent or sabotage is also within scope. The NRC may wish to remove security from the scope of this hearing, but NEPA demands that the possible failure of those systems or programs, such as security, and the environmental costs of their failure are within scope. The voluminous number of security breaches which have occurred at critical infrastructure, including nuclear weapons and power facilities after 9/11 (such as the 16 foreign-born construction workers who were able to gain access to the Y-12 nuclear weapons plant with falsified documentation) demonstrates that nuclear

"insiders" must be deemed potential active participants in an attack. In addition Indian Point is vulnerable to acts of sabotage against off-site power transmission, as was evidenced during the 2003 blackout which struck the Northeast. Various computer systems, at Indian Point, had to be removed from service, including the Critical Function Monitoring System, the Local Area Network, the Safety Assessment System/Emergency Data Display System, the Digital Radiation Monitoring System and the Safety Assessment System.

C) Again, the contents of the DBT, nor the fact that said DBT is a living, constantly changing document, are not the issue nor focus of NEPA and its requirements, but instead what is at issue, is the potential aftermath, if said DBT is found to be inadequate in scope and design.

These three examples are given, as they each would play a part in the aftermath of a terrorist attack at the Indian Point Energy Center located in Buchanan, New York.

6. NEPA's intent and purpose is not in weighing the odds of and event occurring, but instead is intended to measure the risks and costs to the environment should such and event occur. In *San Luis Obispo Mothers*

for Peace v. NRC, 449 F.3d 1016, 1028 (9th Cir. 2006) the courts Memorandum and Order in part states:

> *NRC's "categorical refusal to consider the environmental effects of a terrorist attack" in this licensing proceeding was unreasonable under the National Environmental Policy Act (NEPA).*

It is abundantly clear in the Ninth District Court's ruling, that the odds of a given event are not at issue, but instead the issue is the effects such a postulated event or events would have on the environment. The Ninth Circuit Court Order made it abundantly clear that the NRC must take into consideration the environmental effects of a successful terrorist attack. The NRC had wrongfully attempted to narrow the scope of what will be included within their review based on the NRC's best guess estimates on the odds of such an event occurring.

It is pointed out here, large and small, that there have been 9,438 terrorists events around the world since September 11, 2001. Though most of these attacks were minor in scale and/or thwarted by authorities, the number of attacks speaks volume. The risk of a terrorist attack on a nuclear reactor site is a very real possibility.

NEPA requires the NRC and licensee to answer what are the environmental costs of a successful attack of a terrorist attack on a

Nuclear Reactor site, such postulated events should include, but not be limited to, evaluation of the risks associated with attacking various components of the facility independently and jointly, including for instance the reactor itself, the control room, the spent fuel pools, and the water intake and/or discharge channel, and the attack scenarios should include the attacking force of 9/11, which means scenarios and their aftermaths should include an attacking force of no less than 18 terrorists, the potential use of up to four large commercial airplanes.

Further, attacks should include use of known terrorists weapons of choice which include large vehicle bombs (such as the one used in the Oklahoma City Bombing orchestrated by home grown terrorist Timothy McVay), armor piercing munitions (used for instance by LA gangs and drug cartels), Shoulder launched rockets and grenades, and Semi-Automatic 50 Caliber Rifles (which can be accurate in hitting a target such as a guard tower from up to one mile away, and capable of doing extensive damage from a distance of up to four miles (if successfully hitting a target), and mortars.

> ***Sniper/Anti-Materiel Rifle:*** *53 This weapon was developed by the U.S. military (M82A1) in the 1980s to destroy jeeps, tanks, personnel carriers, and other vehicles. The 28 lb. (12.7 kg) weapon saw extensive use in the Persian Gulf War where a single soldier could disable multiple vehicles in a matter of seconds. It*

fires 50 caliber (0.50 in [1.27 cm] diameter) ammunition and is considered one of the most destructive and powerful weapons legally available in the United States. The price of this weapon can range from $4,000 to $7,000.

This semi-automatic weapon can hit targets accurately one mile (1.60 km) away and can inflict effective damage to targets four miles (6.44 km) away (that is, if the round strikes the target). It can also fire specialized ammunition capable of piercing several inches of metal, exploding on impact, or providing tracers for accurate night shooting. In 1999, GAO investigators noted criminal misuse of 50 caliber weapons in connection with known domestic and international terrorist organizations,

Publicly available sources contain significant weapon capability information:

– U.S. Army's Field Manual FM 3-06.11 [B-1], Combined Arms Operations in Urban Terrain . Chapter 7 of this document is particularly useful and contains weapon penetration information. A wide selection of Army Field Manuals are publicly available for reference and download at www.adtdl.army.mil .

– The Worldwide Equipment Guide [B-2] serves as an interim guide until the publication of Army Field Manual FM 100-65, Capabilities-Based Opposing Force: Worldwide Equipment Guide is published. The Worldwide Equipment Guide is available for reference or download at www.fas.org/man/dod-101/sys/land/row/weg.pdf.

Rocket Propelled Grenade Launcher: The RPG-7 (which is shown below) is a very simple and functional weapon. It is a shoulder-fired, muzzle-loaded grenade launcher that launches a variety of fin-stabilized, oversized grenades from a 40 mm (1.57 in.) tube. It is effective against fixed emplacements, vehicles like tanks, and personnel. Its capability is dependent upon the type of grenade used. Using antitank grenades, its effective range is 500 m (0.31 mi) when used against a fixed target and 300 m (0.19 mi) when fired at a moving target. Its maximum range is 920 m (0.57 mi), at which point the round self-destructs after its 4.5-second

> flight. The antitank round has a lethal bursting radius of 4 m (13.12 ft) when used on an area target. Using an antipersonnel grenade, the RPG-7 can be effective at 1100 m (0.6835 mi). A trained two-man team can fire 4–6 rounds per minute. The weapon is light enough to be carried and fired by a single individual.

Indian Point is vulnerable to water born attacks and aerial assaults. A meltdown can be triggered even at a scrammed reactor if cooling is obstructed. Water intake is also essential to the proper function of spent fuel pools. Yet at certain nuclear plants, cooling systems may be highly vulnerable. At both Indian Point and Millstone Power Station, in particular, water intake pipes have been identified by engineering experts as exposed and susceptible to waterborne sabotage.

> In March 2005, a joint FBI and Department of Homeland Security assessment stated that commercial airlines are "likely to remain a target and a platform for terrorists" and that "the largely unregulated" area of general aviation (which includes corporate jets, private airplanes, cargo planes, and chartered flights) remains especially vulnerable. The assessment further noted that Al Qaeda has "considered the use of helicopters as an alternative to recruiting operatives for fixed-wing operations," adding that the maneuverability and "non-threatening appearance" of helicopters, even when flying at low altitudes, makes them "attractive targets for use during suicide attacks or as a medium for the spraying of toxins on targets below."

The vulnerability of nuclear power plants to malevolent airborne attack is detailed extensively in the Petition filed by the National Whistleblower Center and Randy Robarge in 2002 pursuant to 10 CFR Sec. 2.206. A number of studies of the issue are also reviewed in Appendix A to these Comments. The particular vulnerability of nuclear spent fuel pools to this kind of attack is detailed in the

January 2003 report of Dr. Gordon Thompson, director of the Institute for Resource and Security Studies entitled "Robust Storage of Spent Nuclear Fuel: A Neglected Issue of Homeland Security" and in the findings of a multi-institution team study led by Frank N. Von Hippel, a physicist and co-director of the Program on Science and Global Security at Princeton University and published in the spring 2003 edition of the Princeton journal Science and Global Security under the title "Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States." It is worthy of note that, even post-9/11, general aviation aircraft have circled or flown closely over commercial nuclear facilities without military interception.

Contention is Within Scope in the License Renewal Process

NRC regulation 10CFR 51.53 which is the implementation and enforcement device created by the NRC to abide by the terms and regulations of NEPA demands that the environmental costs of ALL POTENTIAL AND/OR POSTULATED RISKS associated with a major agency action be considered in a Environmental Impact Statement, and further requires that citizens in the potentially affected community be given a chance to have public input into the process and creation of said EIS.

Further, a recent Ninth District Circuit Court Decision in *San Luis Obispo Mothers for Peace v. NRC*, 449 F.3d 1016, 1028 (9th Cir. 2006) we find guidance on the issue at hand in the courts Memorandum and Order in which they state unequivocally:

> *NRC's "categorical refusal to consider the environmental effects of a terrorist attack" in this licensing proceeding was unreasonable under the National Environmental Policy Act (NEPA).*

These two points should be sufficient to prove that this contention is within scope of the process. However, we go further in pointing out that the NRC has provided its own "in agency precedent" to include the potential effects to the environment should there be a successful terrorist attack on a NRC licensed facility. In the license review of an application from Pa'ina Hawaii, LLC., a Hawaiian-owned company, to build and operate an underwater pool-type commercial irradiator at a location near Honolulu International Airport, the NRC staff decided, of their own accord, to include and review the potential of a terrorist attack on the facility, and the resulting environmental effects should a terrorist attack be successfully launched on said facility during its period as a licensed NRC site.

NRC has both a legal and moral responsibility to treat all Stakeholders in a fair and equal fashion, in all regions of the country. The NRC has established a precedent of including the environmental effects of a terrorist

attack on a Licensee site as a part of the EIS in the license renewal process. A Ninth Circuit Court Decision instructed and ordered the NRC to include as a part of the EIS the environmental effects of a successful terrorist attack.

It is clear from the presentation of facts in this document that said contention is within the scope, and deserving of a closer review by the board.

Contention Raises a Material Issue of Fact or Law

Entergy is of the opinion that they are not required to include as a part of their LRA for IP2 the environmental effects of a successful terrorist attack on the Indian Point facility. NRC have exhibited a great reluctance to abide by the legal responsibilities laid out in NEPA, and the NRC's own regulation 10CFR 51.53, as is witnessed by a review of the 48 LRA's that precede the applications for IP2 LLC and IP3 LLC.

Although the commercial interests of the nuclear industry are of valid concern to nuclear utilities and the NEI; they should not be of concern to the NRC. There is no justification for jeopardizing national security and the health and safety of the public and violating NEPA - even to the smallest degree - to safeguard corporate profits.

The Ninth District Court decision, coupled with the NRC own precedent set in the licensing process for the Irradiation Facility in Hawaii shows there are material issues of both the facts and laws presented in this

contention. The Stakeholders of the host community surrounding Indian Point, hold a very different opinion on these facts than does the NRC. The attacks on our sovereign soil here in New York have shown us, proved to us that a terrorist attack is possible, and worthy of inclusion in the EIS for this license application.

Contention is Supported by Facts and/or Expert Opinion

Intervener has met the minimal requirements of the 10 CFR rules and regulations in presenting this contention in a concise statement of the facts adequate to establish that said contention is entitled to a further and complete review of the issues contained herein. It is pointed out that the rules governing the license renewal process, and hearings lay out some basic criteria that a stakeholder must meet to have a contention accepted for further review:

Section 2.309(f)(v) requires "a concise statement of the alleged facts or expert opinion which support the petitioner's position on the issue and on which the petitioner intends to rely at hearing, together with references to the specific sources and documents on which the H petitioner intends to rely to support its position on the issue."

Additionally, it is pointed out that the rules and regulations dealing with hearings and contentions accepted therein goes further to define specifically

the minimum burden of proof necessary to have a contention accepted for further review and scrutiny:

An Intervener is not required to prove its case at the contention filing stage: "the factual support necessary to show that a genuine dispute exists need not be in affidavit or formal evidentiary form and need not be of the quality as that is necessary to withstand a summary disposition motion." Statement of Policy on Conduct of Adjudicatory Proceedings, 48 N.R.C. 18, 22 n.1 (1998), citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process, Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989). Rather, petitioner must make "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate." In Gulf States Utilities Co., 40 NRC 43, 51 (1994), citing, Rules of Practice for Domestic Licensing Proceedings – Procedural Changes in the Hearing Process, Final Rule, 54 F.R. 33168, 33171 (Aug. 11, 1989).

It is clear here, that this contention more than meets the minimal standards necessary for acceptance of this contention. The petitioner in this case has made "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate."

Contention Raises a Material Matter of Fact or Law

1. NRC and PG&E refused to consider the effects on the environment in the case of a successful terrorist attack on the proposed Spent Fuel Facility at Diablo Canyon.
2. Mother's For Peace successfully litigated, and the Ninth Circuit Court handed down a Memorandum and Order that effectively and concreted established law stating that review of the environmental effects in the case of a terrorist attack are to be included in the EIS in a licensing procedure and/application.
3. NRC subsequently implemented a rewrite of the EIS in that licensing review to include (however inadequately) a review of the issues.
4. NRC set agency precedent when it voluntarily included the environmental effects of a possible terrorist attack in the EIS for the licensing of a irradiator facility in Hawaii.
5. FUSE, and the Stakeholders of the host community claim that NEPA's intent is clear, and that all possible risks and incidents and their potential effects on the environment must be reviewed and included in the scope and creation of the EIS for the IP2 LLC LAR.

The NRC cannot approved the Applicant's LRA because it does not address the realistic environmental risk posed by a terrorists attack. The Stakeholders

have raised a material matter of fact or law, thus meeting the burden for further review.

V. CONCLUSION AND REQUEST FOR RELIEF

Entergy's application should be denied by the NRC of the reasons stated above. Alternatively, FUSE seeks protection of its interests through an Atomic Safety Licensing Board (ASLB) Order requiring, as pre-requisite to issuance of new superseding licenses, that Entergy cure the inadequacies in its

application as described above so as to provide assurance of public health and safety. Further, FUSE requests that the Board order that, if and when Entergy curses the inadequacies in its application, Entergy shall then resubmit the relevant portions of its application with appropriate notice and opportunity for adjudication by the ASLB and the parties.

September 21, 2007 Friends United for Sustainable Energy, USA, Inc.

by: _____
Susan Shapiro, Esq.

Attorney for Friends United for Sustainable Energy, USA, Inc.

The undersigned, living within 20 miles of Indian Point 2, has read the above PETITION FOR LEAVE TO INTERVENE, REQUEST FOR HEARING, AND CONTENTIONS, submitted by Friends Untied for Sustainable Energy (FUSE), dated September 21, 2007, and individually join in said Petition, and incorporates the contents of said Petition and all other filed by any stakeholder in this license renew process as if fully set forth herein.

_____
Sherwood Martinelli
351 Dyckman Street
Peekskill, New York 10566
Distance from Indian Point: 2.5

_____
Witnessed and sworn to before me this 21st day of September, 2007

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

We, the undersigned, living within 20 miles of Indian Point 2, have read the above PETITION FOR LEAVE TO INTERVENE, REQUEST FOR HEARING, AND CONTENTIONS, submitted by Friends Untied for Sustainable Energy (FUSE), dated September 21, 2007, and individually join in said Petition, and incorporate the contents of said Petition as if fully set forth herein.

Robert Jones
124 Trails End
New City, 10956
Distance from Indian Point: 8.5

Witnessed and sworn to before me this 21rst day of September, 2007

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 20 11

Susan Hillary Shapiro
36 Horne Tooke Road
Palisades, NY 10964
Distance from Indian Point: 17 miles

Patricia E. French
Witnessed and sworn to before me this 21rst day of September, 2007

PATRICIA E. FRENCH
Notary Public, State of New York
No. 01FR5041486
Qualified in Rockland County
Commission Expires 04/03/07 11

Dr. Sonya Shapiro
Sonya Shapiro
34 Scenic Drive
Suffern, NY 10901
Distance from Indian Point: 9 miles

Witnessed and sworn to before me this 21rst day of September, 2007

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 20 11

Judy Allen
24 Seifert Lane
Putnam Valley, NY 10579
Distance from Indian Point 15 miles

Witnessed and sworn to before me this 21rst day September, 2007

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466

EXHIBIT A

original

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket No. 50-247**
License Renewal Application	)	

DECLARATION OF SUSAN H. SHAPIRO, Esq.

My name is Susan H. Shapiro I live at 36 Horne Tooke Road, Palisades NY 10964, less than 17 miles from Indian Point. I am President of Friends United for Sustainable Energy, USA, Inc (FUSE), a steering committee member of the Indian Point Safe Energy Coalition, and a board member of Hudson River Sloop Clearwater.

I am a life long resident of Rockland County, as is my father and grandfather. I enjoyed walking and cycling along the banks of the Hudson with my family. I am a mother of two children ages 8 and 10.

On 9/11 I first became concerned about the threat of terrorism to Indian Point, as the 9/11 hijackers flew directly over the plant and in fact, had plans to attack before they decided instead to attack the World Trade Center.

As I learned more about the operation of Indian Point, my concern about a terrorist attack, have been dwarfed by the seemingly endless operating problems and leaks at the aging facility, which the current owner, Entergy bought on or about 9/11.

At one of the first annual assessment meetings I attended in Buchanan, NY I was shocked to hear about the amount of repairs required When I expressed my concern an Entergy employee said "it's like an old car, we just keep patching it and it keeps on running". This was my introduction into the lack adequate aging management at the plants.

Since that time, there have been a series of chronic problems with the plant. In 2005 leaks of tritium where discovered accidently near spent fuel pool #2, further investigation uncovered large amounts of Strontium 90 apparently leaking from spent fuel #1. However, to date, the exact location, size,

duration and methods of stopping and remediating said leaks remains unknown.

Other leaks seems to be sprouting up, and are being discovered only by accident, instead of through proper and thorough investigation.

I cannot understand how the NRC can possibly justify issuing Entergy a new superceding license to an additional 20 years, when the plant has clearly outlived its ability to be run without jeopardizing public health and safety, and the integrity of the environment.

The NRC's overly close relationship with the NEI, the nuclear industry's lobby group, became apparent to me, when during a conference in Washington, D.C., during Katrina, the NEI introduced a white paper that reduced the evacuation area guidance from the 10 mile radius, to a 2 mile wedge. NRC quickly rubberstamped favoring protection of the financial profits of the nuclear industry to those of public health and safety, as required by it's organizing mandate.

Indian Point is unique, as it is the only plant located in the middle of 21 million residents, 24 miles from New York City, 3 miles from West Point Military Academy, is leaking Strontium 90, tritium and cesium into the groundwater and Hudson River, and does not have an adequate, workable or fixable evacuation plan, an

Our elected officials, Federal, State and Local, and thousands of Hudson Valley residents have called for Indian Point closure and for an Independent Safety Assessment prior to consideration for relicensing. In fact, even though the NRC refused to require backup power for an emergency siren system, a Federal law was passed that did require such a system be installed and operable months ago. To date Entergy has been unable to properly install the required siren system.

I am aware that the plant is currently leaking dangerous radioactive contaminants from the plant into the ground around the plant, as well as the In the event the river continues to be contaminated from releases from Indian Point, my enjoyment of the river for recreation and exercise will be directly affected. In addition, if Rockland County needs to start using the river for our public water supply my health and the health of my children may be adversely affected.

Indian Point needs to be shutdown, I understand the law requires the site to cleaned up to the condition it was in prior to the plant being built. It appears that the law is being broken. For example in the case of Unit 1, which was shut down over 30 years ago, its spent fuel pool is currently leaking Strontium 90, tritium and cesium into the river. The river is continuing to be polluted as a result of the inaction of the owners and regulators.

If this was any other kind of business, such as a gas station, the government authorities would shut it down and make the owners remediate the underground leaks immediately.

Today, Indian Point could not be sited where it is located in the most densely populated region of the country, on a earthquake fault, and along with the inadequate aging management of the plant, the NRC cannot issue a new superceding license to the operator for another 20 years. In fact the plant should be closed immediately and the cite decommissioned.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 12 day of September, 2007, at Spring Valley, NY.

Susan H. Shapiro, Esq.

Sworn to before me this
12th day of September, 2007.

Patricia E. French

PATRICIA E. FRENCH
Notary Public, State of New York
No. 01FR5041486
Qualified in Rockland County
Commission Expires 04/03/07
11

EXHIBIT B

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	LicenseNo.
Entergy Nuclear Operations	)	DPR-26
Indian Point Energy Center Unit 2	)	Docket
		No. 50-247
License Renewal Application	)	

DECLARATION OF SHERWOOD MARTINELLI

My name is Sherwood Martinelli, and I reside with my wife and seven cats at 351 Dyckman Street in Peekskill, New York. The sinister twin domes of Indian Point are less than three miles from my residence, and can be viewed from the window of our attic sitting room when the leaves are off the trees. I am the founder of the Green Nuclear Butterfly, and Vice President of FUSE USA (Friends United for Sustainable Energy USA, INC. I have resided in Peekskill since the year 2000, my wife since 1996. As a landscaper, I spend a great deal of time interacting with a wide array of local citizens and business owners in Dutchess, Westchester and Rockland counties. My political involvements, including work as a volunteer for Congressman John Halls campaign have me out and visible in the community, my network of friends and business leaders a true pulse of our community. My recreational pursuits see me using the Hudson River for a place to walk, a place to boat (including canoe and kayak) as well as a place where I on occasion fish.

My hobbies include camping, hiking, biking, photography, and walking. A walk along the banks of the Hudson River down by the Peekskill Train Station is an enjoyable way to get some exercise, stretch my legs on a summers day, or chase away winters cabin fever after a Nor'easter has blown on out to sea and the snow melted away.

My concerns with the nuclear industry, and the NRC's inability to adequately police their licensees, date back to my years living in the foothills of South Eastern Ohio which lead up into the Blue Ridge Mountains. I founded the Save The Wills Creek Water Resources Committee when I uncovered six hundred thousand tons of Low Level Radioactive Waste that had been ILLEGALLY dumped into a wetlands that drained into my community's only drinking water supply. Imagine my shock and horror when I learned through investigation that the NRC had failed to police their licensee, Shieldalloy and their predecessors, for a period of over two decades.

I opposed the privatization of the two DOE Gaseous Diffusion Plants (in Portsmouth, Ohio and Paducah, Kentucky, and their transfer to a newly created company USEC which was given three hundred million in tax payer dollars as start up funds. More importantly, I opposed the transfer of jurisdictional control to the NRC, partially because of their deplorable and lack of enforcement.

When I first arrived in Peekskill, I was newly in love, and unaware that two aging leaking nuclear reactors were my neighbors, threatened my life, health and emotional well being. My first inkling came one day while out tending my flowers and the wailing sirens sounded the alarm. Entergy and the NRC are not really good at giving us warning when the tests take place, and half the time when we do manage too know about them the alarms fail in one sense or another. Then came the whisperings from neighbors of an evacuation plan that would not work which got me to begin taking an interest in the Indian Point site.

September 11th, 2001 was, or should have been, a wake up call to all of us. Sadly it did not seem to be, as the NRC seems far more intent in protecting their licensees than human health and the environment.

On September 11th, I was taking my wife to work at Bronx Community College when the report came in over the radio that a plane had crashed into one of the Twin Towers. We walked into the office to see other members of the staff huddled around a small black and white television set, just in time to see the second plane crash into the second tower. In unaminous shock, we all gasped knowing that our lives had forever changed and that America was under attack. I openly wept when the first tower crashed to the ground, and intuitively knew the second would soon follow suit.

The task of shutting down the college began, and duty interceded as my wife and others began implementing the necessary steps to close down and evacuate the campus. Out on the quad a stunned professor walked towards me with tears in his eyes. I came to learn that his son worked in one of the towers, and he could not reach him on the phone. Students were in a daze; some sat on benches crying alone, while others sat with friends, all with numb expressions.

Mid afternoon my wife and I headed home, an odd deathly pall hanging in the air. It was eerie driving up 87, and then on the Sprain. The only cars on the road were emergency vehicles from near and far, headed South into New York City. Four days later, still glued to the TV, exhaustion finally taking over I passed out curled up on the sofa in our living room. Now, six years to the day, I sit here in front of my laptop watching newsreels from the tragic day, and tears still sting my eyes as I write this declaration.

Entergy wants to relicense two aging nuclear relics for 20 more years of operation, telling us at every chance the Indian Point reactors are vital, safe and secure.

Strontium 90, Tritium and Cesium 137 leaks, sporadically working sirens, and sleeping guards speak volumes that drown out Entergy's pathetic lies. The NRC tells us the relicense of the facilities is an acceptable risk in the name of a greater societal good, and that the Emergency Evacuation Plan is adequate, but not up for discussion, and not considered within scope for the license renewal. If we bring up security, or the aftermath of a successful terrorist attack, we are scolded like children. We are told that the security at the plants is the best in the industry (which is not saying much), and that the odds of a terrorist attack are so small as to be almost non existent. This is why the DBT is off limits, and why the NRC contends that they are not required to consider the environmental costs associated with a terrorist attack in their Environmental Impact Statement.

Some look at my long hair, hear my anti-nuclear rhetoric and label me a lunatic. Others shrug their shoulders as if silently accepting their fate. I on the other hand have read, researched and found the lies that are the nuclear industry. Enforcement is not issuing exemption after exemption to your licensees that allow them to ignore rules and regulations created to keep us safe. Issuing Generic Letters alerting licensees of known serious equipment failure, yet requiring little or no mandatory action on the part of their licensees is not regulatory control. Inspections conducted by NRC staff are of no use and provide no incentive to abide by the rules, when every violation is written up as a green non-cited violation, even if the violation presents a serious risk to human health and the environment. A licensing process that lets licensees skirt the requirements of 10 CFR 54 by agreeing to a future list of commitments that they will file a letter requesting relief from is nothing more than a jury rigged system, a deceitful rape of communities being forced, with no real say, to continue hosting what are unsafe and dangerous facilities.

As a citizen who lives three miles from Indian Point, I should not be forced to live in fear, yet the continued operation of Indian Point leaves myself and 21 million other people living within 50 miles of the aging, embrittled reactors any other choice. Basic commonsense tells us that the evacuation plan is a farce. The Witt Report proves it beyond a shadow of a doubt. Rather than be truthful about this fact, Entergy and the NRC are now trying to reprogram public expectations, sell us on Sheltering in Place. Go to the Centers For Disease Control website, and you soon realize that Sheltering in Place is just another con game. The average citizen sheltered in a wood frame or brick home with a concrete basement is only afforded a 40 percent level of protection in the event of a nuclear attack. Eliminate the basement, and that level of protection drops to just ten percent. Entergy tells us it would be only days, and the NRC agrees. Problem is, the State Departments official website tells the real truth, and suggests we be prepared to shelter in place for as long as three weeks. The Department of Homeland Security harkens us to "be prepared," meaningless instructions evidently borrowed from the Boy Scouts.

If the Indian Point reactor and entire nuclear industry is so safe, why can't we as home owners get insurance to cover our losses should a nuclear incident or terrorist

attack occur? Could it have something to do with the overly hardened and brittle reactor cores that are now overly suspect to become victims of thermal shock, or the insurance industry's inside knowledge of what the devastation would be in the case of a successful terrorist attack on a nuclear reactor such as Indian Point? Is it fair to indemnify and hold the nuclear plant harmless in the case of and accident caused by industry and NRC negligence with the recent renewal of the Price Anderson Act?

. When Indian Point Two and Three were cited and built we were promised closed cooling systems, and even an 80 acre forested park with walking paths on the 235 acre site. More than 30 years later, we are still waiting on the closed water cooling system, and the park is just another broken promise. Acceptable risk, and what this risk encompasses should be a community decision, not an agency decision. If 50 percent plus one of us feels the risk of having Indian Point as a neighbor has become too great a risk, then the time to shut the reactors down is upon us. The NRC and Entergy will not accept the reality, but the time has come. Entergy's License Renewal Application should be denied to protect my health and safety, the health and safety of my family, and the health and safety of 21 million people who live within the 50 mile circle of death surrounding Indian Point in the event of a significant nuclear event.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 11th day of September, 2007, at Spring Valley, NY.

Sherwood Martinelli
Vice President FUSE USA
Distance from Indian Point: 8.5
Personal Contact information
351 Dyckman Street
Peekskill, New York 10566

914 734 1955

Witnessed and sworn to before me this 21rst day of September, 2007

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 20

EXHIBIT C

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket**
License Renewal Application		**No. 50-247**

DECLARATION OF JULIE GOTTESMAN

My name is Julie Gottesman I live at 128 Highmount Avenue, Nyack, NY 10960. I am a member of Friends United for Sustainable Energy, USA, Inc (FUSE).

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in Nyack for approximately 6 years. I enjoyed walking and cycling along the river at Hook Mountain with my family. I am a mother of two children under 12 years old. I am currently a member of a rowing team, that practices in the Hudson River 2 to 3 times a week.

As a resident of Rockland County I am concerned that due to the limited water supply, the county is currently considering using the Hudson for drinking water.

I am aware that the plant is currently leaking dangerous radioactive contaminants from the plant into the ground around the plant, as well as the Hudson River.

In the event the river continues to be contaminated from releases from Indian Point, my enjoyment of the river for recreation and exercise will be directly affected. In addition, if Rockland County needs to start using the river for our public water supply my health and the health of my children may be adversely affected.

Indian Point needs to be shutdown, I understand the law requires the site to cleaned up to the condition it was in prior to the plant being built. It appears

that the law is being broken. For example in the case of Unit 1, which was shut down over 30 years ago, its spent fuel pool is currently leaking Strontium 90, tritium and cesium into the river. The river is continuing to be polluted as a result of the inaction of the owners and regulators.

How can the NRC allow the operators continue operating a plant in this condition, let alone consider relicensing it for another 20 years?

If this was any other kind of business, such as a gas station, the government authorities would shut it down and make the owners remediate the leaks immediately.

Many of my neighbors and friends share this view, and are astonished at the apparent inability for the federal government to recognize this obvious lack of oversight to protect my health and my children's health.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 12th day of September 2007, at Nyack, NY.

Julie Gottesman

Sworn to before me this
12th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT D

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket No. 50-247**
License Renewal Application	)	

DECLARATION OF GARY SHAW

My name is Gary Shaw, I live at 9 Van Cortlandt Place, Croton on Hudson, NY 10520. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE), Croton Close Indian Point (CrotonCIP), a member of the Steering Committee of the Indian Point Safe Energy Coalition (IPSEC.

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in the Hudson Valley for 15 years, and proximity to the Hudson was very important in our decision to move to Croton. The Hudson is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be cited in Buchanan, NY, due the enormous surrounding population and lack of a viable evacuation plan. The evacuation plan has been evaluated by a preeminent expert in emergency planning, James Lee Witt, and was judge inadequate and to a large degree, unfixable.

I am involved in Hudson River activities such as many Earth Day riverbank clean-ups during which I have often gotten abrasions and cuts while removing debris from the riverbanks. I have never before been concerned about my activities when pulling illegally dumped debris from the riverbank. Among the materials I have personally extracted are construction materials such as panels of house siding and aluminum window frame, car parts, tires and household appliances such as air conditioner and a refrigerator.

I am aware that the plant is allowed to discharge regulated amounts of radioactive elements into the river and that there are also currently unregulated leaks of radioactive contaminants from an undetermined number of sources into the ground around the plant, and that the contaminated water's pathway is generally towards the Hudson River. With the leakage continuing unabated and the potential for increased flow due to system degradation over time, my participation in river cleanups would have to be reevaluated.

Because this leakage is not yet directly linked to a known source of drinking water, the NRC has declared that the uncontrolled leaks are not a threat to public health or safety. As a user, but not a drinker of the river, I am concerned.

The NRC is considering granting the plant a license renewal that will result in twenty more years of high level nuclear wastes that will also go into spent fuel storage that is leaking now and will continue to leak long after the plant has finally been decommissioned. I am concerned that my health may be compromised because Indian Point currently is and will apparently be allowed to leak radioactivity indefinitely. In fact, since the spent fuel pool at Indian Point 1 is believed to be among the sources of leakage, and Indian Point 1 has been inactive for decades, it appears that the plant will leak into perpetuity. That would appear to be a preview of the future of Indian Point 2. Allowing 20 more years of additional wastes to be generated and stored in leaking pools seems to me to be a direct threat to citizens' health and safety.

If this were any other kind of business, such as a gas station, wouldn't the EPA or other regulatory agency shut it down and make the owners remediate the leaks immediately?

It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and the Hudson River. I am certain that many of my neighbors and friends share this view, as evidenced by the wide-spread willingness to sign petitions in opposition to Indian Point at each year's Croton Village Summerfest, and actions by the village Board of Trustees, including passing resolutions supporting the congressional call for an Independent Safety Assessment, and previously calling for plant closure and opposition to relicensing.

The public's health and safety should not be compromised for the financial benefit of a privately owned corporate polluter, whose parent company has allowed the bankruptcy of another of its nuclear plants in order to avoid financial liabilities in the aftermath of Hurricane Katrina. In addition, many articles have suggested that energy efficiency and conservation programs, and the upgrading of the deteriorating transmission lines would mitigate the perceived need for Indian Point's electrical output.

I declare that the statements made in this declaration are true and correct to the best of my knowledge.

Executed this 15th day of September, 2007, at Croton on Hudson, NY.

Gary Shaw

Sworn to before me this
15th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT E

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket No. 50-247**
License Renewal Application	)	

DECLARATION OF ANDREW Y. STEWART, PhD

My name is Andrew Y. Stewart, I live at 19 Mill Street, Nyack, NY 10960. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE) and the Executive Director of Keep Rockland Beautiful, Inc.

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in the Hudson Valley for 17 years, and am very connected to the Hudson River. About 12 years ago I built my own kayak and use it in the Hudson. Also I have built my own small sail boat.

I teach environmental science at Rockland Community College. For the past 6 years I have organized volunteer clean-up of the banks of the Hudson. For the past 3 years I have helped Hudson River Basin Watch put together educational workshops for high school students on the Haverstraw water front, regarding land use and water quality.

Rockland County is currently considering using the river for tap water, due the limited water resources in the county.

The Hudson River is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be cited where it is currently located, due the enormous surrounding population and lack of a workable evacuation plan.

Indian Point is currently leaking radioactive waste into the groundwater and River, yet the NRC is considering to permit it to continue operating and leaking for another 20 years.

It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and Hudson.

If the NRC permits Entergy to continue operation of this aging plant that is polluting the River, it will directly affect my lifestyle by preventing me from enjoying the river for exercise and will stop me from being able to bring students and community members to its banks. In addition it may directly affect the health and safety of my family.

The public's health and safety cannot be compromised, for the sole benefit of a privately owned corporation.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 10 day of September, 2007, at Nyack, NY.

Andrew Y. Stewart

Andrew Y. Stewart

Sworn to before me this
10th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 20

EXHIBIT F

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, LLC	)	**LicenseNo. DPR-26**
And Entergy Nuclear Operations, Inc.	)	**Docket**
Indian Point Energy Center Unit 2		**No. 50-247**
License Renewal Application	)	

DECLARATION OF TIMOTHY ENGLERT

My name is Timothy Englert. I live at 260 Mirth Drive, Valley Cottage, NY 10989, with my wife and two young children, within in 10 miles of Indian Point. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE) and work for the Palisades Interstate Parks Commission.

I have lived in the Hudson Valley for 7 years, and in both my private and professional life am very involved with the Hudson River. I am an avid kayaker, canoeist and crew rower, and am on the water a minimum of one day a week from the spring through the fall. I use the Hudson river from Bear Mountain down through Piermont.

The Hudson River and Hudson Valley is incredibly beautiful and is vital resource to our community and the entire New York region. As Development Specialist for the Palisades Interstate Park Commission, I am tasked with creating recreational, educational, and philanthropic opportunities with our parks, many of which lie directly on the Hudson River

I try to avoid going near Indian Point when I am kayaking because of the thermal pollution and growing concern about the radioactive leaks into the river.

I understand that the owners of Indian Point, Entergy, have applied for a new license for 20 more years. I cannot see how the NRC can possibly approve this, when every other day there is some problem reported in our local

newspapers about Indian Point, including radioactive leaks, fires, and siren problems.

The population in the Hudson Valley is extremely dense and the road infrastructure is very limited. An accident on the New York State Thruway or the Tappan Zee Bridge causes the traffic to stop for hours. If something happened at Indian Point, evacuation from this area would be nearly impossible.

I understand that Rockland County is currently considering using the river for tap water, due the limited water resources in the county. I am concerned that the ongoing leaks into the Hudson River could have health affects on me and my family.

It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and Hudson river now, let alone for another 20 years.

If the NRC permits Entergy to continue operation of this aging plant that is polluting the River, it will directly affect my lifestyle and that of my children, especially as they embrace the waters of the Hudson.

The public's health and safety cannot further be compromised, for the benefit of a privately owned corporation.

Friends United for Sustainable Energy USA, Inc., (FUSE) represents me in the above cited Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 19th day of September, 2007, at Valley Cottage, NY.

Timothy Englert

Timothy Englert

Sworn to before me this
19th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County

NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket**
License Renewal Application		**No. 50-247**

DECLARATION OF JEANNE SHAW

My name is Jeanne Shaw, I live at 9 Van Cortlandt Place, Croton on Hudson, NY 10520. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE).

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in the Hudson Valley for 15 years. I am an artist who uses driftwood from the Hudson River for many of my pieces. I have spent and continue to spend much time walking along the banks of the Hudson collecting materials for my work.

I am aware that the Indian Point Nuclear Power Plant is currently leaking dangerous radioactive contaminants into the ground around the plant, and that the general flow of the contamination is towards and into the Hudson River. While publicized testing and off-site readings indicate that my beachcombing is currently uncompromised and my art materials are contaminant free, I am concerned that continuing leakage, especially if the aging process leads to faster or larger leaks, will affect to continue my work and interfere with my access to river materials.

I believe the law requires industrial sites to be cleaned up and restored to the condition they were in prior to the plant being built. It appears that the law is being ignored, since Indian Point Unit 1, which was shut down over 30 years ago, is currently leaking Strontium 90, tritium and cesium into the surrounding environment and subsequently, into the river. The river is continuing to be polluted as a result of the inaction of the owners and regulators.

How can the NRC allow the operators to continue operating a plant in this condition, let alone consider relicensing it for another 20 years? If this were any other type of industrial or business site, such as a gas station with leaking tanks or a dry cleaner allowing toxic chemicals to enter the environment, wouldn't either state or federal regulatory authorities shut it down and make the owners remediate the leaks immediately? It is unacceptable for the NRC to allow Indian Point to continue to contaminate the groundwater and the ultimately the Hudson River.

The public's health and safety should not be compromised for the financial benefit of a privately owned corporation. I believe that many of my neighbors and friends share the view that tacit acceptance of radioactive leaks by the federal government's regulators represents a very limited perspective of what constitutes a threat to public health and safety.

I declare that all statements in this declaration are true and correct to the best of my knowledge.

Executed this 15th day of September, 2007, at Croton on Hudson, NY.

Jeanne D Shaw

Jeanne Shaw

Sworn to before me this
15th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT H

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**LicenseNo. DPR-26**
Indian Point Energy Center Unit 2	)	**Docket**
License Renewal Application		**No. 50-247**

DECLARATION OF ROBERT A. JONES

My name is Robert A. Jones, I live at 124 Trails End, New City 10956, with my wife and my three young children. I am a member of Friends United for Sustainable Energy USA, Inc. (FUSE).

FUSE represents my interests in a Petition for Leave to Intervene, Request for Hearing and Contentions; and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc, License Renewal Application.

I have lived in Rockland for 38 years. Until a few years ago I used the river for swimming and water skiing, off my boat. I stopped due to the condition of the water and all the leaks you hear about from Indian Point.

When I was swimming and waterskiing, my friends and I would park our boats just north of the Haverstraw Bay, and we noticed the dramatic different in water temperature. It was always much warmer there. When we learned that is was warmed because Indian Point was dumping heated water into the river we immediately using it, and it turned me off from swimming any where in the river. Indian Point has changed my quality of life.

Now that I know that strontium and tritium in to the river I am even more concerned.

I heard that they are considering using the Hudson River for Rockland County tap water, I think its crazy. Certainly if I won't swim it I wont' drink it or bathe in it. Or permit my young children to do so. This will certainly affect my quality of life.

I love the Hudson River because it is beautiful area, it is close to home, convenient, unfortunately because of Indian Point there are many things I used to do on the water, that I now cannot do.

drink it or bathe in it. Or permit my young children to do so. This will certainly affect my quality of life.

I love the Hudson River because it is beautiful area, it is close to home, convenient, unfortunately because of Indian Point there are many things I used to do on the water, that I now cannot do.

The Hudson River is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be cited where it is currently located, due the enormous surrounding population and lack of a workable evacuation plan.

I work for a company that owns a gas station, where a spill was reported, the DEC and Health Department immediately shut down the station, until it was totally dug up and remediated, even though it turned out not to be the gas stations fault. I cannot understand how our government allows Indian Point to remain open and be considered for relicensing for another 20 years, with all the leaks and problems that keeps arising.

Indian Point is currently leaking radioactive waste into the groundwater and River, yet the NRC is considering to permit it to continue operating and leaking for another 20 years, to me this totally insane.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 12 day of September, 2007, at Spring Valley, NY.

Robert A. Jones

Sworn to before me this
12th day of September, 2007.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2011

EXHIBIT I

NOT LOCATED

LAW OFFICES OF
LEBOEUF, LAMB, LEIBY & MACRAE
1821 JEFFERSON PLACE, N.W.
WASHINGTON, D.C. 20036

October 15, 1968

Dr. Peter A. Morris
Director
Division of Reactor Licensing
U.S. Atomic Energy Commission
Washington, D. C. 20545

Re: AEC Docket No. 50-247

Dear Doctor Morris:

Transmitted herewith are three (3) originals and nineteen (19) conformed copies of Amendment No. 9 to the Application for Licences in the above-captioned proceeding, together with seventy-three (73) copies of the technical material referred to therein.

Copies of these documents will be served today upon Mr. William J. Burke, Mayor, Villiage of Buchanan, New York, and a Certificate of Service will be filed with the Commission later today.

Sincerely yours,

LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Consolidated
Edison Company of New York, Inc.

Enclosures

RECEIVED
OCT 15 1968

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of Consolidated Edison Company of New York, Inc.	))))	Docket No. 50-247

Amendment No. 9

to

Application for Licenses

Consolidated Edison Company of New York, Inc., Applicant in the above-captioned proceeding, hereby files Amendment No. 9 to its Application for Licenses for the purpose of transmitting its Final Facility Description and Safety Analysis Report, consisting of four volumes.

WHEREFORE, Applicant prays as in its original Application for Licenses.

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

By _____
W. Denham Crawford
Administrative Vice President

Dated: October 11, 1968

Subscribed and sworn to before me
this 11th day of October, 1968.

_____
Notary Public

My Commission Expires March 30, 1969.

FRANCES E. FLYNN
Notary Public, State of New York
[illegible]
Qualified in Kings County
Cert. Filed in New York County
Commission Expires March [illegible], [illegible]

1.3 GENERAL DESIGN CRITERIA

The general design criteria define or describe safety objectives and approaches incorporated in the design of this plant. These general design criteria, tabulated explicity in the pertinent systems section in this report, comprise the proposed Atomic Industrial Forum versions of the criteria issued for comment by the AEC on July 10, 1967. The remainder of this section, 1.3, presents a brief description of related plant features which are provided to meet the design objectives reflected in the criteria. The description is developed more fully in those succeeding sections of the report indicated by the references.

The parenthetical numbers following the section headings indicate the numbers of its related proposed General Design Criterion (GDC).

1.3.1 OVERALL PLANT REQUIREMENTS (GDC 1-GDC 5)

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to reactor operation or safety are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected and erected and the materials selected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance.

1.3 GENERAL DESIGN CRITERIA

The general design criteria define or describe safety objectives and approaches incorporated in the design of this plant. These general design criteria, tabulated explicity in the pertinent systems section in this report, comprise the proposed Atomic Industrial Forum versions of the criteria issued for comment by the AEC on July 10, 1967. The remainder of this section, 1.3, presents a brief description of related plant features which are provided to meet the design objectives reflected in the criteria. The description is developed more fully in those succeeding sections of the report indicated by the references.

The parenthetical numbers following the section headings indicate the numbers of its related proposed General Design Criterion (GDC).

1.3.1 OVERALL PLANT REQUIREMENTS (GDC 1-GDC 5)

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to reactor operation or safety are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected and erected and the materials selected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance.

EXHIBIT J

Mrs. Beck

DOCKET NUMBER PR-50
PROPOSED RULE
General Design Criteria

ATOMIC INDUSTRIAL FORUM INC.

850 THIRD AVENUE · NEW YORK, N.Y. 10022 · PLAZA 4-1075

October 2, 1967

DOCKETED
USAEC
OCT 4 1967
Office of the Secretary
Public Proceedings
Branch

Secretary
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Sir:

Pursuant to notice which appeared in the Federal Register of July 11, 1967, the Forum Committee on Reactor Safety is pleased to forward the enclosed comments on AEC's proposed "General Design Criteria for Nuclear Power Plant Construction Permits".

These comments, which in a number of instances take the form of a redraft of the proposed criteria, are based on information developed during an August 9 meeting of the Committee. They have been further refined by a Committee task force comprised of the following members: Wallace Behnke of Commonwealth Edison Company; Arthur C. Gehr of Isham, Lincoln & Beale; R. J. McWhorter of General Electric Company; J. E. Tribble of Yankee Atomic Electric Company; Robert A. Wiesemann of Westinghouse Electric Corporation; and Edwin A. Wiggin of the Forum staff.

The comments have subsequently been circulated to those additional members of the Committee who participated in the August 9 meeting. It may, therefore, be concluded that the enclosed comments generally represent the views of the following additional Committee members:

R. H. Bielecki, Pennsylvania Power & Light Company
Warren S. Brown, Dilworth, Secord, Meagher & Associates, Ltd.
Harvey F. Brush, Bechtel Corporation
Robert W. Davies, Baltimore Gas and Electric Company
William S. Farmer, Allis-Chalmers Manufacturing Company
George C. Freeman, Jr., Hunton, Williams, Gay, Powell & Gibson
Robert E. Kettner, Consumers Power Company
R. W. Kupp, S. M. Stoller Associates
C. A. Larson, Consolidated Edison Company of New York, Inc.
Zelvin Levine, Hittman Associates, Inc.
James V. Neely, Jersey Central Power and Light Company
H. C. Ott, Ebasco Services, Inc.
Joseph W. Ray, Battelle Memorial Institute
Glenn A. Reed, Wisconsin Electric Power Company
Marlin Remley, Atomics International, Inc.
Royce J. Rickert, Combustion Engineering, Inc.

C/8

W. N. Thomas, Virginia Electric and Power Company
Robert E. Wascher, The Babcock & Wilcox Company
Samuel Zwickler, Burns & Roe, Inc.

Although these comments have been throughly reviewed by those individuals listed above, it should be understood that they do not necessarily represent a unanimity of opinion on all the criteria. Members of the Committee who participated in the August 9 discussion, particularly those who find themselves at variance with the views expressed herein, have been urged to make their views known directly to the AEC in behalf of their own respective companies and organizations.

Perhaps a further note of explanation on the enclosed comments is in order.

In the Committee's opinion, the proposed criteria are appreciably better organized than those initially suggested in November 1965. We have also noted with appreciation that some of the Committee's suggestions on the earlier criteria have been accommodated in the criteria now proposed.

The Committee believes that the principal objectives of the criteria should be to assist in the design of nuclear power plants, the preparation of applications for construction permits and operating licenses therefor and regulatory review of these applications to determine if such plants can be constructed and operated without undue risk to the health and safety of the public. The Committee further believes that these objectives should be explicitly stated and that they can be most effectively attained by writing the criteria to the extent possible as performance specifications.

We recommend that the following paragraph be added to the introduction - possibly following the last paragraph of the introduction as it appeared in the Federal Register notice:

> "Each of the requirements stated and implied in the criteria is premised on assuring that the nuclear power plant will be designed, constructed and operated in such a manner as not to cause undue risk to the health and safety of the public from radiation or the release of radioactive materials. To facilitate compliance with the requirements contained in the criteria, the criteria are presented to the extent possible, as performance specifications."

The Committee further believes that the introduction to the criteria should make more explicit reference to their intended direct applicability to water reactors in contrast to their only indirect applicability to reactors of other types, including fast breeders.

Some members of the Committee have noted the desirability and advantages of publishing these criteria as a guide rather than as an appendix to 10 CFR 50. They point out that, as a guide, their interpretation, application and refinement could be more easily adapted to a rapidly

If questions arise in reviewing these comments, the members of the task force would be pleased to meet with representatives of the AEC regulatory staff.

Sincerely,

Edwin A. Wiggin

Edwin A. Wiggin
Committee Secretary

EAW:epb
Enclosure

PROPOSED RULE 50
General Design Criteria

DOCKETED
USAEC
OCT 4 1967
Office of the Secretary
Public Proceedings
Branch

Comments of Forum Committee on Reactor Safety
on
AEC's Proposed Construction Permit Criteria

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

> In the first sentence we have modified "accidents" with "nuclear" and substituted the phrase "cause undue risk to the health and safety of the public" to more precisely reflect what we believe was the AEC's intent. In the last sentence of the original draft, we have dropped the word "sufficiency" since we do not believe that it should be the responsibility of the applicant to document this unless the sufficiency of some specific item is in question. If for any reason the AEC questions the adequacy or sufficiency of a code or standard, it should take this matter up with the appropriate code drafting committee. Note that we have added a sentence requiring a showing of adequacy where there is no applicable code. The balance of the suggested changes are editorial in nature.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear

accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

> The changes in the first sentence are in line with those suggested for Criterion 1. We have deleted the word "additional" on the premise that it is not reasonable to ask the applicant to consider the simultaneous or cumulative forces of more than one extraordinary natural phenomenon.

CRITERION 3 - FIRE PROTECTION (Category A)

A reactor facility shall be designed such that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

> These changes are consistent with the objective of assuring that there will be no undue risk to the health and safety of the public.

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.

As originally drafted, this criterion made unacceptable any impairment of safety, whether the impairment was significant or insignificant. This is unreasonable. Some impairment will undoubtedly result from almost any sharing but the impairment may not be significant enough to preclude the sharing. The test should be whether the sharing will result in undue risk to the health and safety of the public.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

Some of the records that should be maintained may or may not be under the physical control of the licensee or operator. He can, however, assure that they are maintained, by contractual arrangements, if necessary. Those records which are important are those which could have some bearing on the health and safety of the public.

CRITERION 6 - REACTOR CORE DESIGN (Categories A & B)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

We assume that "acceptable fuel damage limits" will be based on "undue risk to the health and safety of the public", not on economic grounds. The latter consideration is a matter for the licensee to decide. Further, these limits will depend on the circumstances leading to the damage. The example "transient situations" have been deleted since they may not be applicable in certain cases and they might also tend to prejudice design innovations.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could

cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

See comment on Criterion 6 with respect to "acceptable fuel damage limits".

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

We recommend deletion of this criterion since it is not applicable to certain reactor types. It is possible for the overall power coefficient resulting from a sum of components with different time constants to be positive without causing any serious safety problem. For example, in a sodium graphite reactor the coefficient has a prompt negative component together with a positive component with a long time constant. This results in an overall positive coefficient, but the negative part of the coefficient is large enough and fast enough to assure satisfactory control and safety. Safety problems relating to reactivity considerations are adequately covered in Criteria 6 and 7.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

It is important to characterize the leakage as "uncontrolled". Our only other suggested change is insertion of the word, "fabricated".

CRITERION 10 - REACTOR CONTAINMENT (Category A)

Reactor containment shall be provided. The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

To avoid any ambiguity, "containment" should be characterized as "reactor containment". The statutory requirement of the licensee and the AEC is "to avoid undue risk to the health and safety of the public", not "to protect the public". It would

be helpful to cross reference this criterion to Criterion 37 to indicate what the AEC means by "engineered safety features". Consistent with our comments on Criterion 37, we have substituted "pipe" for "boundary" on the premise that an applicant should not be required to consider a design basis accident more conservative than the instantaneous double-ended, circumferential rupture of a large coolant pipe.

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.

As originally drafted, this criterion could be interpreted as requiring a second control room. Not only would such a requirement be inconsistent with current practice, we believe that the complexities introduced could adversely affect overall plant safety. We believe it possible to design and equip a control room to assure continuous occupancy under all circumstances, including fire. We have deleted reference to 10 CFR 20 since the radiation exposure limits set forth therein apply to normal operating conditions, not accident conditions. Compliance with the radiation exposure limits of 10 CFR 20 under accident or post-accident circumstances is neither necessary nor reasonable. We have deleted the last sentence of the original draft since it is unnecessary and contradictory with the requirement of continuous occupancy of the control room.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

We have modified this criterion to more accurately and precisely reflect its intent.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

We have dropped the two examples since they are measures of reactivity rather than the fission process.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

We have deleted the phrase "act automatically" since manual action will prove adequate, indeed desirable, in some instances.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

No change suggested.

CRITERION 16 - MONITORING REACTOR COOLANT LEAKAGE (Category B)

Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

We have assumed the intent of this criterion is to assure that leakage from the primary system will be detected, not that the entire reactor coolant pressure boundary will be monitored. The latter requirement would be inconsistent with current practice and unnecessary. Also, consistent with Criterion 9, we believe that the leakage should be characterized as significant and uncontrolled.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.

We believe that the modified language as indicated above more accurately and precisely reflects the intent of the criterion.

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

The suggested change is in line with our comment on Criterion 1.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

The significant change we have made here is to delete the last sentence of the original draft. It would appear preferable to provide duplicates of the best system or component rather than going to an inferior system or component based on a different principle.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

We recommend deletion of this criterion since it is more of a definition than a criterion and since the implied requirement is adequately covered by Criterion 23.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

This criterion should be deleted inasmuch as its requirements, to the extent they should be included in general criteria,

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis.

The suggested change here includes adding to the criterion the phrase, "or shall be tolerable on some other basis".

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

We recommend deletion of this criterion since it would appear preferable to focus all requirements for emergency power in Criterion 39. Note that "protection systems" has been incorporated in Criterion 39 to accommodate this deletion.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

The reason for the changes here is that the licensee should be given some latitude in determining when and how such tests should be carried out. Further, he should be required only to test the active components of a protection system in contrast, for example, to a rupture diaphragm which could only be tested at the expense of destroying it. Also, certain tests might permit the licensee to determine if failure or loss of redundancy has occurred, but they might not permit him to demonstrate it.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

No change suggested.

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

Two independent reactivity control systems, preferably of different principles, shall be provided.

The phrase, "At least" which prefaced the original criterion suggests a possible escalation of requirements which we do not believe was intended

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

> Deletion of the preface phrase, "At least two of" is based on the comment made on Criterion 27. We have deleted the examples at the end of the original criterion since they could be interpreted to indicate a requirement for two fast reactivity shutdown mechanisms. This requirement is unnecessary when there is sufficient redundancy in one of the reactivity control systems to assure shutdown.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

> Deletion of the preface phrase, "At least", is consistent with the comments on Criteria 27 & 28. The other editorial changes are for purposes of clarification.

CRITERION 30 - REACTIVITY HOLDOWN CAPABILITY (Category B)

The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

> Deletion of the preface phrase, "At least one of", is consistent with the comments on Criteria 27, 28 & 29. Further, the public health and safety will not be compromised by a return to low power.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod by

limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

> We believe the criterion should preserve its original objective and at the same time acknowledge that one of the functions of the reactor protection system is to protect against certain control system malfunctions.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

> We believe substitution of "reasonable" for "considerable" and the substitution of "lose capability of cooling the core" for "impair the effectiveness of emergency core cooling" more precisely reflects the intent of the criterion. The re-wording also correctly implies that emergency core cooling will generally be required only if the reactor coolant pressure boundary is breached.

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

> We have deleted the phrase, "and with only limited allowance for energy absorption through plastic deformation", on the premise that it is not helpful.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration shall be given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The detailed requirements contained in the original version are not appropriate for general criteria.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

With the re-writing of Criterion 34 as indicated above, this criterion can and should be deleted.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

It should not be necessary to inspect or maintain surveillance over all portions of the coolant pressure boundary; hence, we have inserted the phrase, "of critical areas". We believe that both the applicant and the AEC are in a better position to take advantage of developing technology and code refinement if these general design criteria refer to "current applicable codes" rather than to specifically designated codes.

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

> Deletion of the phrase, "As a minimum", and substitution of "piping" for "pressure boundary" are both intended to eliminate the implication that the applicant should be required to consider a design accident basis more conservative than the instantaneous, double-ended, circumferential rupture of the largest pipe in the primary system. On this premise, retention of the original language introduces a vagueness which tends to defeat the objective of the criterion.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

> Avoiding undue risk to the health and safety of the public is the purpose of all engineered safety features and the "functional reliability and ready testability" of such features is directly related to their attainment of this objective. To tie this criterion to the problem of siting appears extraneous and not helpful; hence, we have deleted the second sentence.

CRITERION 39 - EMERGENCY POWER (Category A)

An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.

> As originally drafted, this criterion could be interpreted as requiring two off-site and two on-site power sources. Since neither the AEC nor the licensee may have any control over

the off-site power supply and since an emergency on-site power supply adequate to meet the power needs of the engineered safety features is required, any reference to off-site power is irrelevant. We have, therefore, re-written this criterion to eliminate such reference to off-site power. We have also changed the title of the criterion to accommodate the addition of "protection systems", which reference was added because of the deletion of Criterion 24.

CRITERION 40 - MISSILE PROTECTION (Category A)

Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

The suggested changes in this criterion are for purposes of clarification.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

We believe the measure of "sufficient performance capability" of an engineered safety feature should be that no undue risk to the public health and safety will result from the failure of any single active component of that feature. The modified language, in our opinion, more accurately and precisely reflects the intent of the criterion.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.

Although it would appear extremely difficult, if not impossible, to design engineered safety features in such a way that a loss-of-coolant accident will cause no impairment of the capability of any component or system, it is possible to design them to meet the requirements of this criterion as stated above.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided.

> The intent here was simply to state the criterion in a more positive way.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEM CAPABILITY (Category A)

An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

> In our opinion, one emergency core cooling system which incorporates a sufficient redundancy of active components and covers the full range of postulated breaks should be adequate. Our modification of this criterion reflects this consensus. For this reason, we have omitted the last sentence of the original criterion.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM (Category A)

Design provisions shall where practical be made to facilitate physical inspection of all critical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles.

> Since inspection of water injection nozzles is not always possible on a reasonably complete and non-destructive basis and since the failure of a safety injection nozzle is assumed in most accident analyses, we have inserted the phrase, "where practical".

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEM COMPONENTS (Category A)

> No comment other than the criterion should be presented in the context of a single emergency core cooling system, consistent with the comments offered on Criterion 44.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical.

> Testing the "operability" in contrast to the "delivery capability" of the emergency core cooling system "up to" rather than "at" a location close to the core more accurately reflects the art of the possible and should provide for as adequate a test of reliability.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEM (Category A)

A capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources.

> The only change here, and a significant one we believe, is insertion of the word, "initially". Although we concur that a capability to test the operational sequence of the emergency core cooling system should be provided, the test as a practical matter would not be carried out frequently and possibly not more than once - prior to startup.

CRITERION 49 - REACTOR CONTAINMENT DESIGN BASIS (Category A)

The reactor containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public.

> The objective of this criterion, in our opinion, should be that under the circumstances of an accident the integrity of the containment should be such as to prevent

Undue risk to the health and safety of the public. Since the maintenance of containment integrity is based on effective functioning of the emergency core cooling system, it appears unreasonable in this criterion to assume the complete failure of the emergency core cooling system; hence we have assumed a failure of a single active component. Consistent with this assumption, we believe that the pressure and temperature to be withstood should be characteristic of those anticipated from the largest credible energy release associated with a loss-of-coolant accident, including the calculated energy from metal-water and other chemical reactions. Acceptance of the "failure of a single active component" concept is consistent with Criterion 41.

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

The selection and use of containment materials shall be in accordance with applicable engineering codes.

It appears to us that the specific requirements of this criterion as originally drafted are not in keeping with the intent of general design criteria.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, features shall be provided to avoid undue risk to the health and safety of the public in case of an accidental rupture in that part.

It is our understanding that it is the responsibility of the licensee to "avoid undue risk to" rather than "to protect" the health and safety of the public. We have deleted the second sentence of the criterion as originally drafted on the premise that it is only incidental to the requirement set forth in the first sentence.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component.

Deletion of the phrase "at least" is consistent with our comment on Criterion 27. The other changes are consistent with our comments on Criterion 41.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

No change suggested.

CRITERION 54 - INITIAL LEAKAGE RATE TESTING OF CONTAINMENT (Category A)

Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.

> We have inserted "initial" in the title to differentiate Criterion 54 from Criterion 55. Further, we believe it more realistic to leak test at peak pressures associated with postulated accidents than at design pressure. Correlation of leakage rate tests at postulated accident pressures with those conducted at design pressure prior to installation of containment penetrations will permit extrapolation of observed leakage rates to design pressure conditions.

CRITERION 55 - PERIODIC CONTAINMENT LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime.

> Our suggested changes here are consistent with our comments on Criterion 54. Further, a requirement calling for periodic leak testing at design pressure would impose an unnecessary and impractical design requirement on the plant.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.

> We have inserted the word, "periodically" to avoid an interpretation that we do not believe was intended, namely a requirement for "continuous" testing. The other suggested change is consistent with our comments on Criteria 54 & 55.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

> Our only suggested change here is insertion of "to the extent practical". We believe this is consistent with the intent of the criterion as originally drafted, but we also believe that the qualification should be explicit rather than implicit. This comment also applies to Criteria 58, 59, 60, 62, 63, 64 and 65.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

> See comment on Criterion 57.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

> See comment on Criterion 57.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to the extent practical to test periodically the operability of the containment spray system at a position as close to the spray nozzles as is practical.

> Insertion of the phrase, "to the extent practical" is consistent with our comment on Criterion 57. The basis for substitution of "operability" for "delivery capability" is the same as that used in our comments on Criterion 47.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 57.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

See comment on Criterion 57.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 57.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

See comment on Criterion 61.

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

No change suggested.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public.

We have substituted "which would result in undue risk to the health and safety of the public" for "to plant operating areas or the public environs" since we believe the first phrase more accurately describes the responsibility of the licensee.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.

The suggested change permits the criterion to accommodate radiation limits as may be specified which may differ from those set forth in 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.

We have avoided the use of the word, "containment" because of its possible ambiguous connotation. The licensee may rely on some means other than containment to meet the requirements of the criterion. The other suggested changes are consistent with our comments on Criterion 67.

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.

We have deleted the qualification on condition (b) namely, "except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents". This qualification is not helpful and could be subject to misinterpretation by the uninformed public.

basis that the core reactor internals remain functional and that adequate shut down margin can be achieved by control rod insertion, we conclude that the stress and deflection limits for the combined blowdown and design basis earthquake loadings provide an adequate margin of safety.

The primary system side of the steam generators, the pressurizer, and the main coolant pump casings, have been designed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition - Summer 1969 Addenda, as Class A vessels. For other Class I pumps, valves, and heat exchangers the inspection program required independent review of (1) the physical and chemical test data for pressure boundary materials, (2) radiographs of valve bodies, valve bonnets and pump casings, and (3) dye-penetrant examinations of heat exchanger tubes and welds. These requirements resulted in fabrication and inspection programs that contain the essential elements of the recently proposed ASME Codes for Nuclear Pumps and Valves. We find the design codes and inspection requirements acceptable.

We have reviewed the information submitted by the applicant with respect to operating limitations on heatup and cooldown of the primary system imposed by the fracture toughness properties of the materials of the Indian Point Unit 2 reactor vessel. Our evaluation was based on a proposed redraft of section NB-2300 Special Materials Testing (Section III ASME Boiler and Pressure

Vessel Code) dated July 28, 1970, which reflects the material testing requirements in a form consistent with the AEC Fracture Toughness Criteria. As a consequence of our evaluation the applicant has agreed to the heatup and cooldown limitation as presented in Section 3.1-B of the Technical Specifications which represents a modification of his initial submittal. On the basis that these limits reflect a very conservative method of defining pressure vessel fracture toughness, we conclude that they are acceptable.

5.3 Coolant Piping

The reactor coolant piping has been designed in accordance with the requirements of the American National Standards Institute (ANSI) B31.1 Code for Power Piping, 1955 Edition, including the requirements of Nuclear Code Cases N-7 and N-10. All welding procedures and operators were qualified to the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. Additional inspection requirements for the reactor coolant piping during fabrication included ultrasonic and dye-penetrant inspection of all pipe welds. Non-destructive examination of valves included radiographic examination of the valve castings and ultrasonic inspection of all forged components. Dye-penetrant surface examination was also performed. With this program, the inspection of the Indian Point Unit 2 reactor coolant piping substantially

meets the requirements of Class 1 systems under the ANSI B31.7 Code for Nuclear Power Piping adopted in 1969. On this basis we have concluded that the design and inspection program for this system is acceptable.

The original seismic design analysis for the Indian Point Unit 2 reactor coolant system utilized only static methods of analysis. Recently, at our request, the applicant completed a rigorous dynamic analysis of this system utilizing both modal-response spectra and model time-history methods of analyses. As with the reactor internals, the combined loading of a concurrent loss-of-coolant accident blowdown and design basis earthquake was not considered in the design of the Indian Point Unit 2 reactor coolant system. However, the applicant recently completed an analysis of the response of the reactor coolant system to be installed in Indian Point Unit 3 for these combined loads. Since the Indian Point Unit 3 and the Indian Point Unit 2 reactor coolant systems are identical, the applicant has used the results of the analysis for Indian Point Unit 3 in conjunction with the material properties for the Indian Point Unit 2 piping, as determined from tests, to determine that the combined seismic and accident loads can be tolerated by the Indian Point Unit 2 reactor coolant system within acceptable stress limits.

Based on our review of the design limits and analytical procedures employed, we find that the design of the Indian Point Unit 2 reactor coolant system is acceptable.

5.4 Other Class I* (Seismic) Piping

At our request the applicant performed additional seismic analysis on other Class I piping. The adequacy of the seismic design of the feedwater lines, pressurizer surge line, and a typical steam line has been confirmed by a dynamic analysis utilizing the modal-response-spectra method. The adequacy of the seismic design of other Class I (Seismic) piping in the plant was determined by performing a dynamic analysis on selected "worst case" systems. Several systems that are the most vulnerable to dynamic excitation because of system flexibility or location in the supporting structure were analyzed and the resulting stresses compared with the stresses determined by the original static analyses. The applicant has concluded that the conservatism of the original static analysis provided adequate margins to accommodate the previously undetermined dynamic effects.

Based on our review of the original static methods employed and the confirmatory evidence obtained from the recent dynamic analyses of the most vulnerable systems, we have concluded that the design of the Class I (Seismic) piping systems in Indian Point Unit 2 is acceptable.

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*See Section 6.1 for definition of Class I structures, systems, and components.

5.5 Inservice Inspection

An inservice inspection program for the reactor coolant system is included in the Technical Specifications. This program follows Section XI of the ASME Code, Rules for Inservice Inspection of the Reactor Coolant System, as closely as practical. The design of the primary system including the capability to remove insulation at selected areas provides an acceptable degree of access for inspection purposes. The applicant also intends to conduct periodic inservice inspections of the primary pump motor flywheels.

The applicant will review the inservice inspection program with us after five years of reactor operation. It may then be modified based on experience gained during these five years. At that time, we will also require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity.

5.6 Missile Protection

We have reviewed the applicant's primary system layout within the containment in terms of the protection afforded the containment liner and Class I (seismic) systems inside the containment from missiles that might be generated as a result of a primary system failure. We have concluded that adequate protection from potential missiles is provided by the system arrangement and surrounding thick circumferential concrete walls and the concrete floors.

The primary pump motor flywheels installed in Indian Point Unit 2 are the same as those in use in other plants. The flywheels are the standard Westinghouse design, fabricated of A 533B steel. On the basis of the use of high grade material, extensive quality control measures, special manufacturing procedures and preservice and inservice surveillance requirements, we have concluded that assurance has been provided that the integrity of the flywheels will be maintained.

5.7 Leak Detection

The reactor coolant pressure boundary leak detection systems for this plant are similar to those we have reviewed and found acceptable for other plants using a Westinghouse nuclear steam supply system. The systems are based upon air particulate monitoring, radiogas monitoring, humidity detection, and containment sump level monitoring. These systems provide an array of instrumentation that is sensitive, redundant, and diverse and that has adequate alarm features. The sensitivity of these systems is consistent with their primary purpose of detecting any leak in the primary system pressure boundary which could be indicative of incipient failure. The Technical Specifications require that two reactor coolant leak detection systems of different principles shall be in operation when the reactor is operated at power. We conclude that the leak detection systems for Indian Point Unit 2 are acceptable.

5.8 Fuel Failure Detection

The fuel element failure detection system will measure delayed neutron activity in one hot leg of the reactor coolant system. The monitor is connected in series with a delay coil to allow a decay time for N^{16} gamma activity (half life of 7.1 seconds) of about 60 seconds before the coolant reaches the detector. This delay reduces gamma ray background and facilitates detector sensitivity. An alarm signal is provided for the channel. We conclude that this system which is inherently faster in response than previous systems reviewed for other reactors is acceptable.

5.9 Vibration Monitoring and Loose Parts Detection

The major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. Vibration analyses for the core support barrel considered inlet flow impingement and turbulent flow. Natural frequency calculations were made to assure that there would be no deleterious response to known excitations such as pump blade passing and driven frequencies. Fuel bundle response to anticipated driving forces has been calculated and determined by tests in the Westinghouse Reactor Evaluation Center.

The vibration monitoring system to be used for the preoperational test program on Indian Point Unit 2 will consist of mechanical gauges to measure gross relative motion between the thermal shield and core barrel, strain gauges on selected guide tubes, and

accelerometers on the upper core plate. We have concluded that the vibration design analyses and the preoperational test program are acceptable.

In the course of our review of the Indian Point Unit 2 application, it has been noted that techniques for the analysis of neutron noise spectra and accelerometer measurements on the lower heads of primary system vessels might be developed to provide a useful method for inservice monitoring of reactor coolant systems to detect changes in the vibration of reactor components or the presence of loose parts. The applicant has stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefullness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

5.10 Conclusion

Based on our review of (1) the codes and standards used for design, (2) the fabrication and inspection procedures, (3) the inservice inspection program, (4) the provisions for missile protection and leak detection, (5) the provision for fuel failure detection, and (6) the provisions for preoperational vibration

testing and the developmental effort for inservice monitoring to detect vibrations and loose parts, we have concluded that the design and inspection procedures for the reactor coolant system for the Indian Point Unit 2 are acceptable.

6.0 CONTAINMENT AND CLASS I (SEISMIC) STRUCTURES

6.1 General Structural Design

The applicant has categorized as Class I (seismic) those structures (e.g., containment structure and primary auxiliary building), and those systems and components (e.g., reactor vessel and internals, emergency core cooling system), whose failure could cause a significant release of radioactivity or that are vital to the safe shutdown of the facility and the removal of decay heat. We have reviewed the applicant's classification of structures, systems, and components and conclude that they have been classified appropriately.

The Class I (seismic) structures at Indian Point Unit 2 are the containment structure, the primary auxiliary building, the control room building, the fuel storage pool, the diesel generator building, and the intake structure and service water screenwell. The major portion of the primary auxiliary building, the fuel storage pool, and the intake structure are of reinforced concrete construction. The control room building, the diesel generator building, the fuel storage building and the non-Class I portions of the primary auxiliary building are constructed of steel framing with composite metal panel siding.

The environmental conditions that were considered in the structural design include the operating basis earthquake (OBE), the design basis earthquake (DBE), the flooding and wind due to

the probable maximum hurricane, and the flooding due to the probable maximum flood. We have concluded that these conditions were used for the design in an acceptable manner.

6.2 Structural Design and Analysis

The Indian Point Unit 2 primary containment has a free volume of 2.6×10^6 cubic feet and a design pressure of 47 psig. The containment structure is a right cylinder (thickness 4.5 ft) with hemispherical dome (thickness 3.5 ft) mounted on a flat (thickness 9 ft) base mat. The reinforced concrete is lined with 1/4 inch minimum thickness welded ASTM A442 grade 60 firebox quality carbon steel plate. The reinforcing bars conform to ASTM A432 specifications. The reinforcing in the cylinder wall is placed in horizontal and vertical directions with added diagonal tangential reinforcing for earthquake resistance. The reinforcing bars conform to ASTM A432 specifications. Cadweld splices are used in 14S and 18S bars.

We have evaluated the pressure transients that might occur in the containment in the event of a loss-of-coolant accident assuming various sizes of primary coolant system breaks. For the range of postulated break sizes up to and including the double-ended severance of the largest reactor coolant pipe, the largest calculated peak containment pressure is 40 psig. The design pressure of the containment exceeds the calculated peak pressure by more than 10% and is acceptable.

The containment is designed to remain within the elastic range for the 0.10g OBE concurrent with the accident and other applicable loads. It is also designed to withstand the 0.15g DBE concurrent with the accident without loss of function.

We and our seismic design consultant, Nathan M. Newmark, are in agreement with the loading combinations and allowable stresses used by the applicant. Stress and strain limits conform to the requirements of ACI 318-63, Part IV-B. The ACI load factors have been replaced by factors suitable for concrete containment structures.

Based on our review of the design of the containment structure and its capability to withstand the predicted pressures from potential accidents, we conclude that the structural design aspects of the containment are acceptable.

In evaluating the capability of the Class I (seismic) structures, systems, and components, to withstand the dynamic loads due to seismic events, our seismic design consultant, Nathan M. Newmark Consultant Engineering Services, considered the geology and nature of the bedrock, design loads and load combinations, the seismic design parameters, and methods of analysis. On the basis of our review and that of our seismic design consultant, we conclude that the Class I (seismic) structures, systems, and components of Indian Point Unit 2 are designed to accommodate all applicable loads and are acceptable. The report of our seismic design consultant is attached as Appendix G.

During our review we noted a limited number of cases where failure of non-Class I (seismic) structures could potentially endanger Class I (seismic) structures and equipment. These included the Indian Point Unit 1 superheater stack and superheater building, the turbine building, and the fuel storage building. In response to our concern, the applicant performed analyses of these structures using a multi-degree of freedom modal dynamic analysis method, to determine the modifications needed to assure that gross structural collapse of these structures would not occur in the event of a DBE. As a result of these analyses, additional seismic reinforcement is being provided for both the superheater building and the turbine building and the Indian Point Unit 1 superheater stack is to be reduced in height by 80 feet. The truncation of the stack is to be accomplished at a convenient time in the next three years and prior to operation of Indian Point Unit 3. We and our seismic design consultant have reviewed the material submitted by the applicant and conclude that the dynamic analyses performed, and the design modifications proposed, are acceptable.

We have reviewed the as-built wind resistance of Class I structures at the Indian Point Unit 2 facility. Analysis indicates that both the containment and reinforced concrete portions of the primary auxiliary building and intake structure can sustain winds in the range of 300 miles per hour. The control building and diesel generator building which are constructed of structural steel with composite metal panel siding, are estimated by the applicant to be capable of sustaining wind loads of up to 160 miles per hour.

Some natural protection from high winds is afforded the control room building and diesel generator building since they are protected by the turbine building to the west, the Indian Point Unit 1 turbine building, superheater building, and containment to the south, the rising hillside to the east, and the containment and rising hillside to the north.

The wind resistance of the Indian Point Unit 1 superheater stack was also considered with respect to preserving the integrity of Indian Point Unit 2. A reduction in stack height of 80 feet coupled with the additional seismic reinforcement of the superheater building (see discussion above) will enable the stack to resist winds with speeds greater than 300 miles per hour.

On the basis of the very low probability for wind speeds greater than 100 miles per hour at the Indian Point site and on the basis of the wind resistance of the Class I (seismic) structures as discussed above, we conclude that Indian Point Unit 2 is adequately protected against high winds.

6.3 Testing and Surveillance

Strength and leakage tests of the containment building will be performed after construction is completed. A 115% overpressure strength test at 54 psig will be conducted and leakage tests will be made at pressures up to 47 psig. As noted in Section 7.3 of this evaluation, pressurized test channels are provided at all liner seams for long-term surveillance. No permanent instrumentation

is being installed on the containment for strength testing, although examinations will be made for cracking and distortion during the pressure test. Periodic leakage rate tests will be performed on the containment and its penetrations.

We have concluded that the provisions for testing and surveillance of the containment are acceptable. Test and surveillance requirements are included in the Technical Specifications.

6.4 Missile Protection

The possibility exists that missiles might be generated in the unlikely event of a failure of the turbine generator. Although the design criteria for Indian Point Unit 2 did not include consideration of protection against missiles resulting from turbine failures, at our request the applicant has assessed the protection available against missiles that might result from a turbine failure at the maximum overspeed condition (133% of rated normal speed). Specific provisions have been added to limit the off-site consequences that could result from a missile failure, and to provide for safe shut down of the unit. These include an alternative cooling water supply for the charging pumps and added missile protection for a potentially vulnerable portion of the auxiliary steam generator feedwater lines. In addition, a second completely independent turbine speed control system has been provided to reduce the probability of a runaway speed condition that might result in a turbine failure. This

system is designed to the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Criteria No. 279 for protection systems. The Technical Specifications require periodic testing of the overspeed devices to assure operability. We conclude that the applicant has made appropriate provisions to reduce the probability of a destructive turbine missile from being generated and affecting Class I (seismic) items.

The Indian Point Unit 2 reactor vessel cavity is designed to protect the containment against missiles that might be produced by postulated failure of the reactor vessel. Failure of the reactor vessel would result in fluid jet-reaction forces in the cavity wall adjacent to the vessel split or crack as well as stress in the cavity wall from a rise in cavity pressure, both of which would result from coolant blowdown. Also reaction forces in the cavity wall and floor might be produced by the impact of missiles generated by pressure vessel failure. By the use of extensive steel reinforcing, the concrete cavity has been designed to resist both fluid jet and missile impact forces that could result from pressure vessel failure by either longitudinal splitting or various modes of circumferential cracking. The cavity is also designed to sustain a fluid pressure rise to 1000 pounds per square

inch. We have reviewed the applicant's analysis and conclude that the cavity as designed provides adequate protection for the containment liner against missiles that might result from a postulated pressure vessel failure.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling System

The principal equipment of the emergency core cooling system consists of (1) three 50% capacity high pressure safety injection pumps, (2) two 100% capacity residual heat removal pumps for low pressure injection and external recirculation, (3) two 100% capacity recirculation pumps for recirculation internal to the containment, (4) one 100% capacity boron injection tank, and (5) four 33-1/3% capacity accumulators. This system provides redundant capability to inject borated cooling water rapidly into the core in the event of a loss-of-coolant accident and to maintain coolant above the level of the core for an indefinite period following the accident.

The applicant's evaluation of the performance of these systems is based on detailed analyses of (1) the hydraulic behavior of the primary coolant system during and subsequent to a loss-of-coolant accident, and (2) the thermal response of the core during the same period. The analytical methods used to predict the hydraulic behavior of the primary coolant system during a loss-of-coolant accident have been improved significantly during the construction period for Indian Point Unit 2. The original analysis presented in Volume 4 of the FFDSAR was performed with the FLASH-1 hydraulics computer program. This program is limited to a three-node

representation of the coolant system. Subsequent to the analysis performed with FLASH-1, Westinghouse developed a new multi-node hydraulics program called SATAN. Using SATAN the coolant system can be represented with as many as 96 nodes. The SATAN calculations provide considerable detail in the system analysis and increased insight into system performance.

At our request, the applicant reevaluated the performance of the emergency core cooling system during a loss-of-coolant accident using the SATAN multi-node hydraulics code. The applicant's analysis is based on the license application power rating of 2758 MWt. For the case of an accident initiated by a double-ended break in the cold leg primary coolant piping, a maximum fuel element clad temperature of 2015°F was predicted. The applicant's investigation of the emergency core cooling system performance for a range of break sizes and locations indicates that the resultant peak temperatures for any other break will be less than those predicted for the double-ended cold leg break. On the basis of our review of the analytical techniques used in this analysis and our experience with similar analytical techniques, we conclude that there is reasonable assurance that the results obtained with these techniques provide a conservative estimate of the performance of the system in the event of a loss-of-coolant accident at Indian Point Unit 2.

We conclude that the emergency core cooling system will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad water reaction to less than 1% of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for cooling is lost and before the clad is so embrittled as to fail upon quenching and (4) reduce the core temperature and then maintain core and coolant temperature levels in a subcooled condition until accident recovery operations can be accomplished.

In summary, we conclude that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident.

The emergency core cooling system design as presently installed at Indian Point Unit 2 was reviewed by the Division of Reactor Licensing during 1967, subsequent to the issuance of the construction permit on October 14, 1966. This system represented a complete redesign, a considerable increase in flow capability, and enhanced performance when compared to the system reviewed for the construction permit. On the basis that the very significantly improved performance of the redesigned emergency core cooling system provides additional assurance for limiting clad temperatures and maintaining a coolable core we concurred with the applicant's decision to remove the reactor pit crucible from the facility design.

7.2 Containment Spray and Cooling Systems

Two independent heat removal systems are provided to control the containment pressure and temperature following a loss-of-coolant accident. Each system, acting alone at its rated capacity, will prevent over-pressurization of the containment structure. The two systems are the containment spray system and the fan cooling system. The design of each is substantially the same as the design of systems provided at the Ginna plant and other licensed plants.

The containment spray system consists of two 50% capacity spray pumps and is sized to limit the containment post-accident pressure to below design pressure. Sodium hydroxide and boric acid are used as additives to the spray solution to remove radioactive iodine which might be present in the containment after an accident. We have reviewed the use of these chemical spray additives in terms of their iodine removal capabilities, and in addition have evaluated the chemical compatibility of the spray solution with other reactor components. As a result of our review, we conclude that the spray system is adequately sized to cool the containment, that the alkaline spray solution will reduce the iodine concentration in the containment atmosphere, and that corrosion of other materials used in the containment does not introduce a safety problem.

The containment fan cooling system provides complete redundancy to the containment spray system for heat removal from the containment atmosphere during post-accident conditions. Five 20% capacity fan

coolers are provided. Since the fan coolers are located within containment, they must be capable of operating in the post-accident environment. Westinghouse has conducted an environmental test program to demonstrate this capability. Our evaluation of these tests, including the heat removal capability of the heat exchangers, and environmental and radiation testing of the fan cooler motors, valve motor operators and electric cabling indicates that these components will function satisfactorily in the accident environment. An iodine-impregnated charcoal filter system has been included with the fan cooler system to remove organic iodine from the post loss-of-coolant containment atmosphere. The charcoal beds are preceded by demisters and high efficiency particulate air (HEPA) filters.

We have evaluated the inorganic and organic iodine removal capability of the charcoal beds on the basis of tests with steam - air mixtures at 100% relative humidity following prolonged flooding of the bed. We conclude that inorganic and organic iodine removal efficiencies of 90% and 10% per pass, respectively, are conservative values that are justified by the available information.

In summary, we have reviewed the containment spray and fan cooling systems in terms of (1) capability to control the containment temperature, (2) capability to remove inorganic and organic iodine,

(3) system and component redundancy, and (4) capability to function in the post-accident containment environment. We conclude that there is reasonable assurance that these systems will operate as proposed subsequent to a loss-of-coolant accident.

7.3 Containment Isolation Systems

In addition to the usual capability of isolating all lines leading to and from the containment, the Indian Point Unit 2 containment is provided with additional systems to minimize the potential leakage of fission products subsequent to an accident. A containment penetration and weld-channel pressurization system provides for continuous pressurization of zones enclosing containment penetrations and the welds in the containment liner. The system continuously maintains an overpressure of clean, dry air that is in excess of the containment design pressure. Pressurized zones include each piping penetration, each electrical penetration, double gasketed spaces on the personnel and equipment hatches, and the channels over weld seams of the containment liner. The air pressure is maintained by the instrument air compressors with backup from the plant air compressors and from a standby source of nitrogen cylinders. Pressure indication and alarm instrumentation is provided locally and in the control room to assure that loss of pressure will be detected and corrected.

In addition, an isolation seal water system has been provided to assure containment isolation by (1) injecting seal water between the seats and stem packing of the globe and double disc isolation valves used on larger lines, and (2) injecting seal water directly into the line between the closed diaphragm valves used in the smaller lines penetrating containment. Seal water injection is provided for all lines connected to the reactor coolant system and for lines that may be exposed to the containment atmosphere subsequent to an accident. Although the use of the seal water system following a loss-of-coolant accident provides an additional means of reducing leakage, we have not considered the effect of this system in determining the offsite radiological consequences.

We have concluded that the capability provided for isolating the containment is acceptable.

7.4 Post-Accident Hydrogen Control System

In the event of a loss-of-coolant accident, radiation from the core and from escaped fission products will dissociate some of the cooling water into gaseous hydrogen and oxygen. Continued evolution of hydrogen would increase the concentration in the containment to a point where ignition could occur and thus provide an additional energy source.

Redundant flame recombiner units are installed within the Indian Point Unit 2 containment. Each unit has the design capability to prevent the ambient containment hydrogen concentration from exceeding two percent by volume. The units are designed to function, following the loss-of-coolant accident in a containment pressure environment of 1 to 5 psig. Each recombiner system consists of (1) a flame recombiner unit located within containment, (2) a control panel located outside of containment, and (3) a hydrogen gas stand located outside of containment. On the basis of (1) our detailed review of the design of the system and its controls, (2) satisfactory performance testing of the device, and (3) satisfactory environmental testing of those portions of the recombiner system installed within the containment, we conclude that there is reasonable assurance that the recombiner system will perform its intended post-accident function.

In addition, the applicant will provide the capability for purging the containment atmosphere through appropriate filters as an alternate backup means of hydrogen control. The containment penetrations to be used for this system are installed. The design and installation of the equipment required will be performed during the first two years of operation at power.

8.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

8.1 Reactor Protection and Control System

The reactor protection system instrumentation for Indian Point Unit 2 is the same as that installed at the Ginna plant. The adequacy of the protection system instrumentation was evaluated by comparison with the Commission's proposed general design criteria published on July 11, 1967, and the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 2 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the primary loops out of service the reactor is normally automatically limited to 60% of full power. However by manual adjustment of several protection system set points in a manner consistent with the Technical Specifications adequate reactor protection can be provided for operation up to 75% of full power.

We have reviewed the applicant's analysis of the seismic response of the protection system instrumentation and associated electrical equipment and find that adequate testing has been performed on the nuclear instrumentation, switch gear, and process system instrumentation.

In connection with our review of potential common mode failures we have recently considered the need for means of preventing common failure modes from negating scram action and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has been responsive to our request for information and has provided the results of analyses which indicate that the consequences of such transients are tolerable for the existing Indian Point Unit 2 design at a power level of 2758 MWt. Although additional study is required of this general question, we conclude that it is acceptable for the Indian Point Unit 2 reactor to operate at a power level of 2758 MWt while final resolution of this matter is made on a reasonable time scale.

8.2 Initiation and Control of Engineered Safety Features

The instrumentation for initiation and control of engineered safety features for the Indian Point Unit 2 is the same as that installed at the Ginna plant. This basic design has been reviewed extensively in the past and we consider it to be acceptable.

We have reviewed the capability for testing engineered safety feature circuits during reactor operation. Resistance tests will be used for routine determinations of the operability of the master and slave relay coils. The circuits upstream of these relays can be partially tested during operation. During plant shutdown, circuits can be tested completely by coincident tripping of instrument channels and a consequent operation of the master and slave relays in the entire downstream initiating system. We have concluded that this

testing capability is acceptable for Indian Point Unit 2.

8.3 Off-Site Power

Two 138 kilovolt (kV) lines connect the Buchanan switchyard to the Millwood switching station, which in turn is connected to the Consolidated Edison grid and the Niagara Mohawk and Connecticut Light and Power systems. Two additional 138 kV lines, using a separate route from the first two lines, connect the switchyard to the Orange and Rockland tie.

The applicant stated that an analysis of the transmission system has indicated that the system is stable for the loss of any generating unit including Indian Point Unit 2.

A single 138 kV line connects the Buchanan switchyard to Indian Point Unit 2. In addition, three 13 kV lines connect the switchyard to Indian Point Unit 1. Three 138/13 kV transformers in the switchyard feed these three 13 kV lines. While the 138 kV system is the normal supply for the auxiliary load associated with plant engineered safety features, one of the three Indian Point Unit 1 13 kV lines is available to provide power via automatic switching to Indian Point Unit 2 through a 13/6.9 kV transformer. By switching circuit breakers in Indian Point Unit 1, the other two 13 kV lines can also be made available to provide power to Indian Point Unit 2. As the 13/6.9 kV supply is not capable of carrying the total plant auxiliary load for Indian Point Unit 2, the main coolant pumps and the circulating water pumps must be tripped off before the supplies are switched.

We conclude that the off-site power supply provides an adequate source of power for the engineered safety features and safe shutdown loads.

8.4 Onsite Power

Onsite power is supplied by three independent diesel generator sets connected in a separate bus configuration such that there is no automatic closure of tie breakers between the three buses to which the generators are connected. The redundant engineered safety feature (ESF) loads are arranged on the three separate buses such that failure of a single bus will not prevent the required ESF performance under accident conditions. The design engineered safety feature and safe shutdown loads per diesel generator are 1813, 2210, and 2353 HP for the first one-half hour following a loss-of-coolant accident. The loads are then changed to 2438, 2235, and 2043 HP for the recirculation phase of the emergency core cooling system operation. On the basis of our evaluation, we have determined that the appropriate diesel generator ratings are 2200 HP continuous, and 2460 HP for 2,000 hours. We note that some of the estimated emergency loads are above the continuous rating of the machines, but below the 2,000 hour ratings. We consider that this margin is acceptable for Indian Point Unit 2.

Each diesel generator is started automatically upon initiation of emergency core cooling system operation or upon under-voltage on its corresponding 480-volt emergency bus. The generators are

housed in a separate Class 1 (seismic) structure. On-site diesel fuel storage capacity provides a minimum of seven days operation at the required safety feature loads. These design and operating features are acceptable for Indian Point Unit 2.

Our review of the ac auxiliary power system has disclosed that there is adequate capacity and an adequate degree of physical and electrical separation of redundant features. The 125 volt dc system consists of two individually housed batteries. The dc system is divided into two buses with a battery and battery charger for each bus. Each of the two station batteries has been sized to carry its expected loads for a period of two hours following a plant trip at a loss of all ac power.

We conclude that the onsite emergency power system is acceptable.

8.5 Cable Installation

We have reviewed the applicant's cable installation relative to the preservation of the independence of redundant channels by means of separation, and relative to the prevention of cable fires through proper cable rating and tray loading. This has been performed by reviewing the cable installation criteria and method of layout design and by field inspection of electrical cable installation during construction.

A single electrical tunnel carries the electrical cables from the electrical penetration area of the containment to the control building. This tunnel carries all of the electrical cables except the power cables for the reactor coolant pumps, the pressurizer

heater cables, and the control rod power cables. The cables in the tunnel are arrayed on either side of a three-foot aisle in trays or ladders. Separation is provided for in the form of distance, metal separators, or transite barriers. The electrical tunnel does not contain any spliced cable connections. Therefore, the probability of a fire is reduced. Further, a fire detection system and an automatically operated water spray system are provided in the tunnel. Tunnel cooling is provided for by redundant cooling fans. On the basis of adequate separation within the tunnel, a minimum number of heat producing cables and features, redundant cooling systems, and fire detection and spray systems we conclude that the single electrical tunnel is acceptable.

Sixty electrical penetrations are provided in a single electrical penetration area to provide for entry of signal, control, and power cables into the containment. The penetrations are located on three-foot centers, both horizontally and vertically, and are of the hermetically sealed type. As a result of our review, fire barriers in the form of transite sheets were added to separate the power cable penetration from the instrument and control cable penetrations. In addition, as a result of our review certain modifications were made to the cabling in the penetration area, including shortening of cable runs and elimination of cable loops. The segregation of power cables and the shortening of the cable runs reduces the probability of failure by fire and on this basis, we consider the single electrical penetration area acceptable for Indian Point Unit 2.

The applicant has performed a design audit to verify the separation of redundant engineered safety feature power and control electrical cabling. A design review of instrument cabling was also performed on a sample basis.

On the basis of our review of cable installation at Indian Point Unit 2, we conclude that the resulting cable layout, as installed, is acceptable.

8.6 Environmental Testing

Westinghouse has conducted an environmental test program for the instrumentation and controls that are located inside containment and that must function in the environment following a loss-of-coolant accident. We have reviewed the results of this testing program and conclude that the essential instrumentation and controls will function properly in the accident environment.

9.0 RADIOACTIVE WASTE CONTROL

Liquid and gaseous waste handling facilities are designed to process waste fluids generated by the plant so that discharge of liquid and gaseous effluents to the environment will be minimized. Liquid waste is processed both by direct removal of radioactive material with ion exchange resins and by evaporative separation. Using these methods the volume of radioactive waste will be greatly concentrated and the purified liquid streams will either be reused or discharged. Small quantities of radioactive liquid waste will be released routinely to the condenser circulating water discharge canal common to all three units where the waste will be diluted and discharged to the Hudson River.

The limits on routine radwaste releases from the three units that are planned for operation at the Indian Point site will require that the combined releases from the three units when added together be within the limits specified in 10 CFR Part 20. This requirement is stated in Section 3.9 of the Technical Specifications for both liquid and gaseous effluents.

The liquid effluent releases from the three nuclear facilities will be discharged from a common discharge canal into the Hudson River. The nearest sources of public drinking water supplies from the Hudson River are located at Chelsea, New York (backup water supply for New York City) and at the Castle Point Veterans Hospital, 22 and 20.5 miles upstream of the Indian Point site, respectively.

During dry periods with low fresh water river flow, tidal action could carry the radioactivity discharge into the river at the Indian Point site upstream to these river water intake points. Conservative analyses made by the applicant indicate that the concentration of radionuclides at these public water intake points would be less than 1% of the concentration of radionuclides being discharged into the river at Indian Point. Since the releases at the site will be less than the limits of 10 CFR Part 20 (and are expected to be less than 10% of the 10 CFR Part 20 limits, based on past experience with Indian Point Unit 1 and other pressurized water reactor plants), the radioactivity levels at these intakes due to the discharges at Indian Point will not be significant.

Gaseous wastes containing some radioactivity are stored in one of four gas decay tanks. One gas tank is utilized for filling, one for holdup for a 45-day decay period, one for discharging to the atmosphere, and one is held in reserve. Disposal of gaseous wastes from Indian Point Unit 2 is by discharge through the plant vent.

The routine gaseous radioactivity releases from the three nuclear facilities will be from three different vents. The combined release of gaseous waste containing radioactivity from these three sources will be limited by the Technical Specifications such that annual average concentrations at the minimum exclusion distance will not exceed the limits of 10 CFR Part 20, Appendix B,

of the Commission's regulations. For gaseous halogens and particulates with half-lives greater than eight days, the applicable limits of the Technical Specifications are less than 1% of the limits given in 10 CFR Part 20. The Technical Specifications also require that the maximum release rate of gaseous waste not exceed the annual average limit.

Based on our review we conclude that the means provided by the applicant for the disposal of radioactive waste are substantially the same as those we have approved for other facilities and are acceptable. We also conclude that acceptable means are provided and will be used to keep the release of radioactivity from the plant within ranges that we consider to be as low as practicable.

10.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown include (1) the chemical and volume control system, (2) the residual heat removal system, (3) the component cooling system, and (4) the service water system. The systems necessary to assure adequate cooling for spent fuel include (1) the spent fuel pool cooling system, (2) the fuel handling system, and (3) the service water system. The designs for these systems are substantially the same as those we reviewed and found acceptable for the Ginna plant.

10.1 Chemical and Volume Control System

The chemical and volume control system (1) adjusts the concentration of boric acid for reactivity control, (2) maintains the proper reactor coolant inventory and water quality for corrosion control, and (3) provides the required seal water flow to the reactor coolant pumps. The amount of boric acid to be added to the core for reactivity control is determined by the operator. The addition of unborated water as a result of operator error could result in an unintentional dilution during refueling, reactor startup, and power operation. The applicant's analysis indicated that because of the slow rate of dilution there is ample time for the operator to become aware of the dilution and to take corrective action. The applicant is actively participating in the development of a device for continuous monitoring of the reactor coolant boron concentration and will evaluate the feasibility of installing such a monitor when developed.

Our review of the chemical and volume control system emphasized those portions involved in routine and emergency injection of concentrated boric acid. We conclude that the design is acceptable.

10.2 Auxiliary Cooling Systems

Subsystems for auxiliary cooling are the component cooling system, the residual heat removal loop, the spent fuel pool cooling loop, and the service water system. The piping for these three systems is designed to the ANSI B31.1 Code for Pressure Piping.

These systems are equivalent in purpose and design to those of other recently licensed plants. On the basis of our review of this plant and others using the similar systems, we have concluded that these systems are acceptable.

10.3 Spent Fuel Storage

The fuel handling system is designed to transfer spent fuel to the storage pool and to provide storage for new fuel. The spent fuel storage facility is basically the same in capacity and design as those used in previously licensed pressurized water reactor plants. The fuel pool is sized to accommodate spent fuel from 1-1/3 core loadings.

As in other designs, mechanical stops will be incorporated in the crane to restrict motion of the spent fuel cask to its assigned area, adjacent to one side of the fuel storage pool. In addition, the spent fuel racks in the area adjacent to the fuel cask storage

location would be used only in the event that a complete core is unloaded and one-third of a core from a previous unloading is already in storage.

The pool floor is located below grade level and founded on solid rock. Structural damage from a dropped fuel cask would not result in a rapid loss of water from the pool. Makeup water can be supplied from the demineralizer water supply at a flow rate of 150 gpm. Additional water can be provided in an emergency by the use of temporary hookups to other sources.

As a consequence of our evaluation of the potential consequences of a postulated fuel handling accident, the applicant has agreed to provide charcoal filters in the refueling building to reduce the calculated offsite doses that might result in the event of a fuel handling accident in the refueling building. The installation of the filters will be completed during the first year of full power operation.

We conclude that the designs of the spent fuel storage pool and the fuel handling system are acceptable.

11.0 ANALYSES OF RADIOLOGICAL CONSEQUENCES FROM DESIGN BASIS ACCIDENTS

11.1 General

In order to assess the safety margins of the plant design, a number of operating transients were considered by the applicant, including rod withdrawal during startup and at power, moderator dilution, loss of coolant flow, loss of electrical load, and loss of ac power. The reactor control and protection system is designed so that corrective action is taken automatically to cope with any of these transients. Based on our evaluation of the information submitted by the applicant and our evaluations of other PWR designs at the operating license stage, we conclude that the Indian Point Unit No. 2 control and protection system design is such that these transients can be terminated without damage to the core or to the reactor coolant boundary, and with no offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including a control rod ejection accident, an accident involving rupture of a gas decay tank, a steamline break accident, a steam generator tube rupture accident, a loss-of-coolant accident, and a refueling accident.

The calculated offsite radiological doses that might result from the control rod ejection accident, and the accident involving rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

The consequences of the steamline break and the steam generator tube rupture accidents can be controlled by limiting the permissible concentrations of radioactivity in the primary and secondary coolant systems. The Technical Specifications for the Indian Point Unit No. 2 facility limit the primary and secondary coolant activity concentrations such that the potential 2-hour doses at the exclusion radius that we calculate for these accidents do not exceed 1.5 Rem to the thyroid or 0.5 Rem to the whole body.

Our evaluations of the loss-of-coolant accident and the refueling accident are discussed in the following sections.

11.2 Loss-of-Coolant Accident

The design basis loss of coolant accident (LOCA) for the Indian Point Unit No. 2 plant is similar to that evaluated for other PWR plants in that a double-ended break in the largest pipe of the reactor coolant system is assumed.

Although the basis for the design of the emergency core cooling system is to limit fission product release from the fuel, in our conservative calculation of the consequences of the LOCA we have assumed that the accident results in the release of the following percentages of the total core fission product inventory from the core: 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens that are released from the core is assumed to plate out onto internal surfaces of the containment

building or onto internal components and is not available for leakage. We assume that 10% of the iodine available for leakage from the containment is in the form of organic iodide, and that 5% is in the form of particulate iodine. The reactor is assumed to have been operating at a power of 3217 MWt prior to the accident. The primary containment is assumed to leak at a constant rate of 0.1 percent of the containment volume per day for the first day and 0.05 percent per day thereafter. We evaluated the iodine removal capability of the sodium hydroxide containment spray system and assumed an inorganic iodine removal constant of 4.5 per hour for the spray system. We evaluated the iodine removal capability of the iodine impregnated charcoal filter system and assumed a removal constant of 0.49 per hour for inorganic iodine and a removal constant of 0.048 per hour for organic iodine. Iodine particulates are assumed to be removed by the high efficiency particulate air filters. The inhalation rate of a person offsite is assumed to be 3.5×10^{-4} cubic meters per second.

For the calculation of the two-hour dose at the site boundary we used an atmospheric dispersion factor corresponding to Pasquill Type "F" stability, with a 1 meter per second wind speed and an appropriate building wake effect. We calculated the potential doses at the site boundary for this 2 hour period to be 180 Rem to the thyroid and 4 Rem to the whole body. At the low population zone boundary our calculated potential doses for a 30-day period are 270 Rem to the thyroid and 7 Rem to the whole body.

In evaluating the above doses, no credit was given for the isolation valve seal water injection system, the penetration pressurization system, or the weld channel pressurization system. Operation of these systems, which interpose a high gas pressure or seal water area between the containment and the outside atmosphere at all points where leakage might occur, should significantly reduce the leakage rate from the containment, and, thus reduce the doses following an accident. These systems are well designed and tested, and should be available in the event of an accident (see Section 7.3). We did not consider the effect of these systems in our dose calculations because it is inherently difficult to accurately measure leakage rates of less than 0.1% per day by current testing methods.

The control room for Indian Point Unit No. 2 was not designed to meet the requirements we have imposed in more recent construction permit reviews, that the dose for the course of the accident to occupants of the control room be limited to 5 Rem to the whole body and 30 Rem to the thyroid. In order to provide additional protection to the control room occupants in the event of a loss-of-coolant accident, the applicant has equipped the control room with protective clothing and self-contained air respirators for the operators. In view of these provisions, we have concluded that the control room, as constructed, is acceptable in this regard.

11.3 Fuel Handling Accident

We have evaluated the potential consequences of a fuel handling accident, in which it is postulated that a fuel assembly is dropped in the spent fuel pool or transfer canal. We assumed that: (1) all 204 rods in the dropped bundle are damaged, (2) the accident occurs 90 hours after shutdown of the core from which the dropped bundle has been removed, (3) 20% of the noble gases and 10% of the iodine in the dropped fuel bundle are released to the refueling water and the dropped fuel bundle has been removed from a region of the core which has been generating 1.43 times the average core power, (4) 90% of the released iodine is retained in the refueling water, (5) the fission products released from the pool are discharged to the atmosphere by the building recirculation system through charcoal filters with an iodine removal efficiency of 90%, and (6) the same meteorological conditions exist as were assumed for the loss-of-coolant accident. The resultant calculated doses at the site boundary are 146 Rem to the thyroid and less than 4 Rem to the whole body.

11.4 Conclusions

We have calculated offsite doses for the design basis accidents that have the greatest potential for offsite consequences using assumptions consistent with those we have used in previous safety reviews of PWR plants and have found the resulting calculated doses to be less than the guideline values of 10 CFR Part 100.

12.0 CONDUCT OF OPERATIONS

12.1 Technical Qualifications

The Indian Point Unit 2 facility was designed and is being built by Westinghouse as prime contractor for the applicant. Preoperational testing of equipment and systems at the site and initial plant operation will be performed by Consolidated Edison personnel under the technical direction of Westinghouse. The applicant's experience in the power production field is largely with thermal power plants. However, the applicant has operated Indian Point Unit 1, a 615 megawatt (thermal) pressurized water reactor plant with an oil fired superheater, since August 1962. In addition, the applicant has the Indian Point Unit 3 under construction at the Indian Point site and is actively considering the installation of other nuclear power plants at other sites. Our review of the applicant's organization indicates that the competence of its engineering staff has continually increased and is consistent with the requirements of its expanded nuclear program.

12.2 Operating Organization and Training

The applicant's organization consists of three main groups under the direction of the general superintendent. These groups are the operations group (with a separate superintendent for each unit), the performance group (with the responsibility for station chemistry, licensed personnel training, and surveillance of station performance),

and the health physics group headed by a supervisor engineer for health physics (with the responsibility for station health physics and instrumentation). An assistant superintendent for maintenance, and production engineers (responsible for providing staff support for the operation superintendents) report to the two superintendents for operation. A reactor engineer reports directly to the general superintendent.

The proposed shift complement for the combined operation of Indian Point Unit 1 and Indian Point Unit 2 consists of one general watch foreman licensed as a senior reactor operator (SRO), one watch foreman (SRO) for each unit, one control operator A licensed as a reactor operator (RO) for each unit, one unlicensed control room operator B, shared by both units, one control operator B for Indian Point Unit 1 chemical system building, six operating mechanics (two of whom are assigned to Indian Point Unit 2), one shift chemist, and one shift health physics technician.

The shift composition for Indian Point Unit 2 when Indian Point Unit 1 is shutdown for any reason is the general foreman, one watch foreman, one control operator A and two operating mechanics. In addition, a control room operator B may be available a substantial portion of his time. We conclude that both the dual unit crews and single unit crews as outlined above are acceptable.

Since a large part of the plant staff has had prior nuclear experience, the training program has been fitted to individual needs based on experience, educational background and job responsibilities. The training program includes long- and short-term assignments of key staff personnel to technical institutions and operating reactors, to the Westinghouse offsite operator training school, and to on-site classroom training courses for operators and supervisors conducted by both applicant and Westinghouse personnel. We have reviewed these activities in detail and conclude that the combination of reactor operating experience and formal training obtained by the plant staff has adequately prepared them to perform their operational duties.

As a means for the continuing review and evaluation of plant operational safety, the applicant will expand the responsibilities of the Nuclear Facility Safety Committee currently functioning for Indian Point Unit 1 to include Indian Point Unit 2. The committee, which reports to the Executive Vice President, Central Operations, will have a membership of at least 12 persons, and will have responsibilities to: (1) audit and report upon the adequacy of all procedures used in the operation, maintenance, and environmental monitoring of each nuclear plant; (2) review and report upon the adequacy of all proposed changes in plant facilities and procedures pertaining to operation, maintenance, and environmental monitoring and having safety significance;

(3) review and report upon all proposed changes to the Technical Specifications; (4) conduct unannounced spot inspections of plant monitoring operations; (5) review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plant; and (6) convene, at the request of the nuclear power generation manager or a nuclear plant general superintendent or chairman or vice chairman of the committee, to review and act upon any matter they may deem necessary.

Westinghouse will participate in the startup and initial operation of the plant and will continue to make available technical support to the Indian Point Unit 2 staff during operation of the facility.

We conclude that the applicant's organization is acceptably staffed and technically qualified to perform its operational duties subject to satisfactory completion of licensing examinations of personnel requiring licenses.

12.3 Emergency Planning

The site emergency plan for the Indian Point site describes the emergency organization and its responsibilities. The scope of the emergency plan includes consideration of local contingencies, site contingencies, general (off-site) contingencies, implementation levels for each contingency, notification channels, the support provided by civil authorities, protective measures for each

contingency, communications facilities, and training drills. The applicant has provided an extensive description of the medical support that will be available although it is not incorporated explicitly in the plan. The planned medical support provides for emergency treatment of plant personnel both at the site and at a designated hospital where facilities equipment and medical personnel to handle radiation contaminated injured personnel will be available.

We conclude that the applicant's emergency plan is acceptable for Indian Point Unit 2.

12.4 Industrial Security

The immediate plant area (restricted area), including Indian Point Unit 1 will be enclosed by a fence. Access to the restricted area for all personnel will be through manned gatehouses or locked gates which are under the direct control of the station security forces. Security guards will make routine patrols of all property within the site boundary and outside the restricted area and are required to make hourly reports to the central control room.

The controlled area of Indian Point Unit 2 will include the containment, the fuel storage building, the primary auxiliary building, and the emergency diesel generator building. Normal access to these areas is through the existing security room for Indian Point Unit 1. All other doors and hatches leading into the controlled area will be locked and will be supervised by means of door switches connected to the open door alarm board in the

security room, and the category alarm board in the Indian Point Unit 1 central control room. The containment personnel hatch doors have remote indicating lights and annunciators that are located in the control room and that indicate the door operational status.

Offsite applicant employees must identify themselves at the main gate prior to admission to the restricted area, receive approval for entry by the general superintendent or his designated representative, and sign in on an admission sheet. If access into the controlled area is approved, they must be accompanied by a qualified guide.

We conclude that the applicant has taken reasonable measures to provide for the security of the facility.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define safety limits and limiting safety system settings, limiting conditions for operation, periodic surveillance requirements, certain design features, and administrative controls for the operating plant. These specifications cannot be changed without prior approval of the AEC. The applicant's initial proposed Technical Specifications, presented in Amendment No. 20, have been modified as a result of our review to describe more definitively the allowable conditions for plant operation. The Technical Specifications as approved by the regulatory staff, may be examined in the Commission's Public Document Room.

Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and that means are provided for keeping the release of radioactivity from the plant within ranges that we consider as low as practicable. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features to mitigate the consequences of unlikely accidents will be available.

14.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS reported on the application for construction of the Indian Point Unit 2 at the proposed site in a letter dated August 16, 1966. The applicant has been responsive to the recommendations made by the ACRS in that letter, and we conclude that the matters raised have been resolved satisfactorily during the design and construction of the Indian Point Unit 2.

The ACRS reported on its review of the application for an operating license for Indian Point Unit 2 in their letter, dated September 23, 1970, attached as Appendix B.

In its letter, the ACRS made several recommendations and noted several items all of which have been considered in the indicated sections of our evaluation. These include: (1) reevaluation of potential flooding at the Indian Point site (Section 3.4), (2) additional seismic reinforcing at the Indian Point Unit No. 1 superheater building and truncation of the superheater stack (Section 6.2), (3) reactor design, power distribution, and control of potential xenon oscillations (Section 4.2), (4) containment design and isolation (Sections 6.2 and 7.3), (5) containment cooling and iodine removal systems (Section 7.2), (6) emergency core cooling system and removal of the reactor pit crucible (Section 7.1), (7) post-accident hydrogen control (Section 7.4),

(8) charcoal filters in the refueling building (Section 10.3), (9) reactor core instrumentation (Section 4.2), (10) reactor protection with only three of four loops in service (Section 8.1), (11) inservice vibration monitoring and loose parts detection (Section 5.9), (12) fuel failure detection (Section 5.9), (13) availability requirements for primary coolant leak detection systems (Section 5.7), (14) pressure vessel fracture toughness (Section 5.2), (15) integrity of high burnup fuel during design transients (Section 4.3), and (16) common mode failure and anticipated transients without reactor scram (Section 8.1).

The ACRS concluded in its letter that if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

15.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and all of the directors and principal officers of the applicant are United States citizens.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes, is involved. For these reasons and in the absence of any information to the contrary, we have found that the activity to be performed will not be inimical to the common defense and security.

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations that relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR Part 50.33(f) and 10 CFR Part 50 Appendix C. The Consolidated Edison Company's application as amended by Amendment No. 21 thereto, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated operating cost for each of the first five years of operation plus the estimated cost of permanent shutdown and maintenance of the facility in a safe condition. The estimated operating costs are $10.0 million for 1971 (the first year of operation), $14.8 million for 1972, $12 million for 1973, $10.9 million for 1974 and $10.7 million for 1975 (Amendment No. 21). Such costs include the costs of operating and maintenance and fuel. The applicant's estimate of the cost of permanently shutting down the facility and maintaining it in a safe condition is (1) $265,000 for the first year of shutdown and $50,000 for each year thereafter if the reactor core is removed from the vessel, and (2) $240,000 per year if the core is not removed.

We have examined the certified financial statements of the Consolidated Edison Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1969 financial report indicates that operating revenues

for 1969 totaled $1,028.3 million; operating expenses (including taxes) was $830.5 million; the interest on the long-term debt was earned 2.3 times; and the net income for the year was $127.2 million, of which $102.1 million was distributed as dividends to the stockholders, and the remainder of $25.1 million was retained for use in the business. As of December 31, 1969, Company's assets totaled $4,069.6 million, most of which was invested in utility plant ($3,793.3 million), and earnings reinvested in the business were $426.1 million. Financial ratios computed from the 1969 statements indicate a sound financial condition, (e.g., long-term debt to total capitalization--0.52, and to net utility plant--0.52; net plant to capitalization--0.994; the operating ratio--0.81; and the rates of return on common--7.7%; on stockholder's investment--6.9%; and on total investment--4.9%). The record of the Company's operations over the past 5 years reflects that operating revenues increased from $840 million in 1965 to $1,028 million in 1969; net income increased from $111.8 million to $127. million; and net investment in utility plant from $3,170 million to $3,793 million. Moody's Investors Service (August 1969 edition) rates the Company's first mortgage bonds as A (high-medium grade). The Company's current Dun and Bradstreet rating (July 1970) is AaAl.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR Part 50.33(f) with respect to the operation of Indian Point Unit 2. A copy of the staff's financial analysis is attached as Appendix H.

17.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities such as power reactors under 10 CFR Part 50.

17.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The Consolidated Edison Company, is with respect to Indian Point Unit 2, subject to the foregoing requirements, and has taken the following steps with respect thereto.

The Company has furnished to the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) Nos. NF-100.

Further, the Company executed Indemnity Agreement No. B-19 with the Commission as of January 12, 1962, which was amended to cover its pertinent preoperational fuel storage under license SNM-1108 on March 4, 1969. The Company has paid the annual indemnity fee applicable to preoperational fuel storage.

17.2 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million.

Accordingly, no license authorizing operation of Indian Point Unit 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended and that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. The amount of financial protection required for a reactor having the rated capacity of this facility would be $82 million. Consolidated Edison Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licensees, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

18.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the Consolidated Edison Company of New York, Inc., dated December 6, 1965, as amended (Amendments Nos. 9 through 25, dated October 15, 1968, October 13, 1969, October 24, 1969, November 21, 1969, December 29, 1969, January 27, 1970, March 2, 1970, March 30, 1970, April 17, 1970, June 3, 1970, July 14, 1970, July 17, 1970, July 28, 1970, July 29, 1970, August 13, 1970, August 28, 1970, and November 12, 1970, respectively) complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. Construction of the Indian Point Nuclear Generating Unit No. 2 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-21, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
5. The applicant is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The applicable provisions of 10 CFR Part 140 have been satisfied; and
7. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to Consolidated Edison for Indian Point Unit No. 2, the Commission's Division of Compliance will prepare and submit a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformaty of this construction to the provisional construction permit and the application. Before an operating license will be issued to Consolidated Edison for Indian Point Unit No. 2, assuming such a license is authorized following the public hearing, the facility must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to license issuance.

APPENDIX A

CHRONOLOGY OF

REGULATORY REVIEW OF THE CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2

(SUBSEQUENT TO CONSTRUCTION PERMIT NO. CPPR-21

ISSUED ON OCTOBER 14, 1966)

1.	April 17, 1967	Submittal of Amendment No. 6 containing design information on the Emergency Core Cooling System and other areas as requested by the ACRS in their letter to the Chairman AEC, of 8/16/66.
2.	July 18, 1967	Meeting with applicant to discuss revised design of Emergency Core Cooling System and other areas as per Amendment No. 6.
3.	August 2, 1967	Letter to applicant requesting additional information on subjects addressed by the ACRS in their letter of 8/16/66.
4.	October 16, 1967	Submittal of Amendment No. 7 in response to DRL request of August 2, 1967.
5.	October 31, 1967	Submittal of Amendment No. 8, revised pages for Amendment No. 7.
6.	December 28, 1967	ACRS Subcommittee meeting to discuss emergency core cooling system, reactor pit crucible, primary coolant system, other areas.
7.	January 30, 1968	Submittal of "Report on the Containment Building Liner Plate Buckle in the Vicinity of the Fuel Transfer Canal".
8.	February 2, 1968	Meeting with applicant to discuss content of Amendments No. 6, 7, and 8.
9.	February 13, 1968	Meeting with applicant to complete discussion of February 2, 1968.

10.	March 8, 1968	ACRS Full Committee meeting to discuss Emergency Core Cooling System; reactor internals; primary coolant system, design, fabrication, in-service inspection, and leak detection; core design; reactor pit crucible; and containment liner quality control and stress analysis.
11.	October 15, 1968	Consolidated Edison Company filed application for an Operating License for the IP-2 Plant. Amendment 9, Volumes 1, 2, 3, & 4.
12.	March 5, 1969	AEC-DRL requested additional information on medical and emergency plans.
13.	March 12, 1969	AEC-DRL staff met with Con Ed personnel to discuss scheduling of regulatory review of application for operating license.
14.	April 3, 1969	AEC-DRL staff met with Con Ed personnel to discuss structural and seismic design and tornado protection.
15.	April 16, 1969	AEC-DRL staff met with Con Ed to discuss accidental and normal radioactivity release from the IP-2 plant.
16.	April 28, 1969	Con Ed requested extension of completion date for construction of the IP-2 plant.
17.	May 2, 1969	AEC-DRL staff and Nathan M. Newmark, seismic design consultant, met with Con Ed personnel at the IP-2 site to discuss seismic design and review status of construction and site inspection.
18.	May 19, 1968	AEC-DRL staff issued an order extending completion date for construction of the IP-2 plant to June 1, 1970.

19.	August 4, 1969	Request to applicant for additional information on site and environment, reactor coolant system, containment system, engineered safety features, instrumentation and control, electrical systems, waste disposal and radiation protection, conduct of operations, and accident analysis.
20.	August 22, 1969	AEC-DRL staff requests copies of monitoring reports and status of actions on Fish and Wildlife recommendations.
21.	August 23, 1969	ACRS Subcommittee meeting on tornado protection, emergency planning, permanent in-core instrumentation, adequacy of onsite emergency power, and containment isolation.
22.	September 24, 1969	Meeting with applicant to discuss Westinghouse presentation on power distribution detection and control in Indian Point 2.
23.	October 13, 1969	Submittal of Amendment 10 (Supplement #1) responses to AEC regulatory staff's request of March 5, 1969, on medical plans and partial answers to AEC regulatory staff's request for additional information of August 4, 1969.
24.	October 24, 1969	Submittal of Amendment No. 11, replacement pages and responses to AEC regulatory staff's request for additional information of August 4, 1969, on Sections 1, 4, 5, 6, 7, 12, and 14 of the FSAR.
25.	November 13, 1969	Request for additional information on reactor, reactor coolant system, containment system, engineered safety features, auxiliary and emergency systems, initial tests and operations, and accident analysis.
26.	November 21, 1969	Submittal of Amendment No. 12, additional and replacement pages to be inserted into the FFDSAR and further responses to AEC regulatory staff's request for additional information of 8/4/69 on Sections 1, 4, 7, 8 and 11 of the FFDSAR.

27.	December 10, 1969	Meeting with applicant to review electrical drawings including AC power, DC power, Reactor Protection System, and Engineered Safety Features.
28.	December 30, 1969	Meeting with applicant and Westinghouse Electric Corporation to continue detailed review of electrical drawings including Reactor Protection System and Engineered Safety Features.
29.	January 16, 1970	Meeting with applicant to review and discuss electrical drawings including Reactor Protection System and Engineered Safety Features.
30.	January 21, 1970	Meeting with applicant & Westinghouse Electrical Corporation on technical specifications.
31.	January 27, 1970	Submittal of Amendment No. 14, replacement pages for FSAR & further responses to AEC-DRL questions of 8/4/69 & 11/13/69, chapters 1, 4, 6, 11, 12 & 14.
32.	February 17, 1970	Meeting with applicant for presentation of results of Con Ed's Analysis concerning potential damage to Indian Point 2 and IP-3 from a failure of the IP-1 superheater stack.
33.	March 2, 1970	Submittal of Amendment No. 15, responses to AEC regulatory staff's requests for additional information of 8/4 and 11/13, 1969 and Containment Design Report.
34.	March 10, 1970	Request to applicant for additional financial data.
35.	March 13, 1970	Meeting with applicant to discuss questions concerning core heat transfer and burnout limits, fuel element performance and ECCS performance during a LOCA.

36.	March 19, 1970	Meeting with applicant, Westinghouse presentation on iodine removal system for IP-2.
37.	March 26, 1970	Meeting with applicant to discuss analysis of fresh water flood and changes to electrical systems.
38.	March 30, 1970	Submittal of Amendment No. 16, additional and replacement pages for the FSAR and further responses to the AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
39.	April 25, 1970	ACRS Subcommittee meeting and meeting with applicant on instrumentation and control, and anticipated transients with failure to scram.
40.	April 17, 1970	Submittal of Amendment No. 17, additional and replacement pages to be inserted into the FSAR and further responses to AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
41.	April 29, 1970	Meeting with applicant to discuss seismic and structural design questions for IP-2.
42.	May 5, 1970	Meeting with applicant to discuss failure mode analysis of the engineered safety feature manual actuation panel.
43.	May 11, 1970	ACRS Subcommittee meeting at the Indian Point 2 site to discuss instrumentation and control and Electrical Systems.
44.	May 12, 1970	AEC issued Order extending completion date for construction of the IP-2 plant to June 1, 1971.
45.	May 28, 1970	ACRS Subcommittee meeting to discuss loss-of-coolant accident, anticipated transients with failure to scram.
46.	June 3, 1970	Submittal of Amendment No. 18, additional and revised pages for the FSAR in response to AEC regulatory staff request for additional information.

47.	June 11, 1970	ACRS full Committee meeting to consider design of engineered safety feature manual actuation panel and operation with less than four loops.
48.	June 17, 1970	Meeting with applicant to discuss consequences of turbine missiles, sensitized stainless steel control room accident dose, hydrogen recombiner.
49.	July 15, 1970	Submittal of Amendment No. 19 (Supplement 10), additional and revised pages for the FSAR and Flooding Evaluation report.
50.	July 20, 1970	Submittal of Amendment No. 20, (Supplement 11) proposed Technical Specifications.
51.	July 24, 1970	Request for additional information on emergency core cooling, reactor coolant system, instrumentation and control, electrical systems, conduct of operations and accident analysis.
52.	July 28, 1970	Submittal of Amendment No. 21, Con Ed Annual Report.
53.	July 28 and 29, 1970	ACRS Subcommittee meeting to discuss technical specifications, flood protection, Unit No. 1 superheater stack failure and containment sprays.
54.	July 30, 1970	Submittal of Amendment No. 22, (Supplement 12), revised pages for FSAR in response to request for additional information.
55.	August 7, 1970	Meeting with applicant to discuss technical specifications.
56.	August 13, 1970	ACRS full Committee meeting to discuss the matters addressed in our July 2, 1970 report.
57.	August 14, 1970	Submittal of Amendment No. 23 (Supplement 13), answers to request for additional information issued July 24.

58.	August 18, 1970	Meeting to discuss licensed operator requirements.
59.	August 28, 1970	Submittal of Amendment No. 24 (Supplement 14). Revised pages to the FSAR.
60.	September 1, 1970	Meeting with applicant regarding performance of Emergency Core Cooling System.
61	September 9, 1970	Meeting with the applicant to discuss Technical Specifications.
62.	October 21, 1970	Request to applicant for a report on analysis of laminations in base plate material of the IP-2 pressurizer.
63.	October 29, 1970	Meeting with applicant to review technical specifications for the Indian Point 2 plant.
64.	November 1970	Submittal of Amendment 25 (Supplement 15), changes to technical specifications and to FSAR.

APPENDIX B

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

SEP 23 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consolidated Edison Company of New York, Inc., for authorization to operate the Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the Committee's 95th, 98th, 122nd, and 124th meetings, and at Subcommittee meetings on August 23, 1969, March 13, 1970, April 25, 1970, May 28, 1970, July 28-29, 1970, and September 15, 1970. Subcommittees also met at the site on December 28, 1967 and May 11, 1970. The Committee last reported on this project to you on August 16, 1966. During the review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Indian Point site is located in Westchester County, New York, approximately 24 miles north of the New York City limits. The minimum radius of the exclusion area for Unit No. 2 is 520 meters and Peekskill, the nearest population center, is approximately one-half mile from the unit. Also at this site are Indian Point Unit 1, which is licensed for operation at 615 MWt, and Unit 3, which is under construction.

The applicant has re-evaluated flooding that could occur at the site in the event of the probable maximum hurricane and flood, in the light of more recent information, and has concluded that adequate protection exists for vital components and services.

Additional seismic reinforcement being provided for the Indian Point Unit No. 1 superheater building and removal of the top 80 ft. of the superheater stack will enable the stack to withstand winds in the range of 300-360 mph corresponding to current tornado design criteria. Since

the reinforcement of the superheater building, which supports the stack, enables the stack to resist wind loads of a magnitude most likely to be experienced from a tornado, the Committee believes that removal of the top 80 ft. of the stack, to enable it to resist the maximum effects from a tornado, may be deferred until a convenient time during the next few years, but prior to the commencement of operation of Indian Point Unit No. 3. The applicant has stated that truncation of the stack will have no significant adverse effect on the environment.

The Indian Point Unit No. 2 is the first of the large, four-loop Westinghouse pressurized water reactors to go into operation, and the proposed power level of 2758 MWt will be the largest of any power reactor licensed to date. The nuclear design of Indian Point Unit No. 2 is similar to that of H. B. Robinson with the exception that the initial fuel rods to be used in Indian Point Unit No. 2 will not be prepressurized. Part-length control rods will be used to shape the axial power distribution and to suppress axial xenon oscillations. The reactor is designed to have a zero or negative moderator coefficient of reactivity, and the applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

Unit 2 has a reinforced concrete containment with an internal steel liner which is provided with facilities for continuous pressurization of weld and penetration areas for leak detection, and a seal-water system to back up piping isolation valves. In the unlikely event of an accident, cooling of the containment is provided by both a containment spray system and an air-recirculation system with fan coolers. Sodium hydroxide additive is used in the containment spray system to remove elemental iodine from the post-accident containment atmosphere. An impregnated charcoal filter is provided to remove organic iodine.

Major changes have been made in the design of the emergency core cooling system as originally proposed at the time of the construction permit review. Four accumulators are provided to accomplish rapid reflooding of the core in the unlikely event of a large pipe break, and redundant pumps are included to maintain long-term core cooling. The applicant has analyzed the efficacy of the emergency core cooling system and concludes that the system will keep the core intact and the peak clad temperature well below the point where zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The Committee believes that there is reasonable assurance that the Indian Point Unit No. 2 emergency core cooling system will perform adequately at the proposed power level.

The Committee concurs with the applicant that the reactor pit crucible, proposed at the time of the construction permit review, is not essential as a safety feature for Indian Point Unit No. 2 and need not be included.

To control the concentration of hydrogen which could build up in the containment following a postulated loss-of-coolant accident, the applicant has provided redundant flame recombiner units within the containment, built to engineered safety feature standards. Provisions are also included for adequate mixing of the atmosphere and for sampling purposes. The capability exists also to attach additional equipment so as to permit controlled purging of the containment atmosphere with iodine filtration. The Committee believes that such equipment should be designed and provided in a manner satisfactory to the Regulatory Staff during the first two years of operation at power.

The applicant plans to install a charcoal filter system in the refueling building to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. This installation will be completed by the end of the first year of full power operation.

The reactor instrumentation includes out-of-core detectors, fuel assembly exit thermocouples, and movable in-core flux monitors. Power distribution measurements will also ordinarily be available from fixed in-core detectors.

The applicant has proposed that a limited number of manual resets of trip points, made deliberately in accordance with explicit procedures, by approved personnel, independently monitored, and with settings to be calibrated and tested, should provide an acceptable basis for the occasional operation of Indian Point Unit No. 2 with only three of the four reactor loops in service. The Committee concurs in this position.

The applicant stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefulness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

The reactor includes a delayed neutron monitor in one hot leg of the reactor coolant system to detect fuel element failure. Suitable operability requirements will be maintained on the several sensitive means of primary system leak detection.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

The applicant stated that existing experimental results and analyses provide considerable assurance that high burnup fuel of the design employed will be able to undergo anticipated transients and power perturbations without a loss of clad integrity. He also described additional experiments and analyses to be performed in the reasonably near future which should provide further assurance in this regard.

The Committee has, in recent reports on other reactors, discussed the need for studies on further means of preventing common failure modes from negating scram action, and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has provided the results of analyses which he believes indicate that the consequences of such transients are tolerable with the existing Indian Point Unit No. 2 design at the proposed power level. Although further study is required of this general question, the Committee believes it acceptable for the Indian Point Unit No. 2 reactor to operate at the proposed power level while final resolution of this matter is made on a reasonable time scale in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept advised.

Other matters relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS letters should, as in the case of other reactors recently reviewed, be dealt with appropriately by the Staff and the applicant in the Indian Point Unit No. 2 as suitable approaches are developed.

The ACRS believes that, if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit No. 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 2

1. Amendment No. 9 to Application of Consolidated Edison Company of New York for Indian Point Nuclear Generating Unit No. 2, consisting of Volumes I - IV, Final Safety Analysis Report, received October 16, 1968

2. Amendments 10 - 20 to the License Application
3. Amendments 22 - 24 to the License Application

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis Report
Volumes I, II, III and IV dated October 15, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
November 29, 1968

As pointed out in our comments of October 29, 1965 on Unit No. 2, a primary influence on the meteorological statistics of the Indian Point site seems to be its location in a river valley about a mile wide with terrain rising 600 to 1000 feet on either side. Consequently, wind directions follow a pronounced diurnal cycle with daytime, unstable (lapse) flow in the upriver direction and nighttime, stable flow in the downriver directions. The report documents a 42.4 percent inversion frequency, but it should also be pointed out that inversion conditions are largely confined to the nighttime, downriver flow lasting about 12 hours before changing to lapse or upriver flow. Figure 2.6-1, although in terms of average vectors, shows the marked wind reversals at sunset and sunrise and the rather persistent, channeled flow that can occur during the middle of the night (see the mean direction between 0200 and 0800 hours). The mean wind speeds during this persistent period is about 2.5 m/sec which indicates that 50 percent of the time inversion wind speeds could be less than 2.5 m/sec.

In the absence of specific, joint-frequency wind speed and direction persistence data from the site, a reasonably conservative meteorological model would be to assume for a ground release a 1 m/sec wind speed under inversion conditions in a persistent downriver direction for a period of 8 hours. Taking into account the likelihood of a diurnal wind reversal, a very conservative assumption would be to allow the plume centerline to meander over a 22-1/2° arc under the same conditions for the remainder of the 24-hour period. Again, with no specific on-site wind persistence data, the conservative assumption has been made.

The amount of additional atmospheric diffusion because of the building turbulence can be assessed by the virtual point source expression $[(x + x_o)/x]^{1.5}$ as used by the applicant, which for a value of x_o = 430 m

amounts to a factor of 2.5 at the site boundary (520 m) and 1.6 at the low population boundary (1100 m). These values are in close agreement with the method of using a shape factor of 1/2 and a building cross-section of 2000 m^2.

In summary, from data presently available, it would seem reasonably conservative to assume a persistent wind direction for an 8-hour period under inversion conditions and a 1 m/sec wind speed. With the added assumption of a building wake shape factor of 1/2 and a cross-sectional area of 2000 m^2, the resulting 0-8 hr relative concentration would be 6.6×10^{-4} sec m^3 at the site boundary and 3.7×10^{-4} at the low population boundary. From Table 14.3.5-3 one can calculate that the applicant's model for the 0-8 hr period results in an average relative concentration of 4.8×10^{-4} and 2.4 sec m^{-3} at the site and low population boundary, respectively.

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis
Amendment No. 12 dated November 21, 1969, and
Amendment No. 14 dated January 27, 1970

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 17, 1970

The original documentation of the Indian Point site during the period 1955-1957 indicates that at the 100-ft. height the annual prevailing wind direction is from the north northeast and that in the sector from 22.5 to 42.5 degrees the frequency of inversion, neutral and lapse conditions was 6, 2, and 1 percent, respectively. Within this sector, the shortest site boundary is approximately in a direct line through Units 2 and 3 at a distance of 610 and 380 m, respectively, as measured from figure 2.2-2. It is about 500 m from the Unit 1 stack to this common boundary point. The nearest site boundary, regardless of sector, is where the property line intersects the downriver edge of the site. Although this point is at a distance of 580 m from Unit 2, it is not in the most prevalent wind direction by a considerable amount.

To compute the average annual dilution factor we have assumed the frequencies listed above, averaged over a 20-degree sector with a wind speed of 2, 4 and 6 m/sec, respectively, for inversion (Type F), neutral (Type D), and lapse (Type B) conditions. Assuming no building wake effect our results show the applicant's values for Units 1 and 2 to be reasonably conservative. In the case of Unit 3 we compute an average annual dilution factor of 2.9×10^{-5} sec m^{-3} as compared to the applicant's value of 1.6×10^{-5} sec m^{-3}. The only explanation we have for the ESSA value being twice as high is the use of the building wake effect in the applicant's assumptions.

It is our view that the use of the building wake effect in the long-term average diffusion equation, as was done by the applicant, is inappropriate. It does not seem logical that for the same atmospheric conditions the Sutton equation on page Q 11.10-1 for the long-term model gives <u>more</u> credit for building wake effect than the equivalent short-term model on p. Q 11.10-2. For example at $x = 400$ m assuming $x_0 = 400$ m and $n = 0.5$, the building wake effect, $[(x+x_0)/x]^{2-n/2}$, for the long-term equation is 3.4 whereas for the effect in the short-term equation, $[(x+x_0)/x]^{2-n}$, the value is 2.8. It is the larger exponent in the former that makes the difference. Also, the fact that one averages in the horizontal dimension over a sector essentially would nullify any added dilution in that dimension because of wake effect.

APPENDIX D

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CERER

21 November 1969

Mr. Roger S. Boyd
Asst. Director for Reactor Projects
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Boyd:

Reference is made to your letters regarding Docket Nos. 50-247, 50-286, 50-342, and 50-343, Consolidated Edison Company of New York's proposed Indian Point Nuclear Generating Units No. 2 and No. 3, and Units No. 4 and No. 5 which are contiguous to Indian Point plant site.

Pursuant with our arrangements, Mr. R. A. Jachowski and Mr. B. R. Bodine of CERC have reviewed all pertinent information contained in the reports from the standpoint of establishment of a design water level. This included the review of the storm surge associated with the Probable Maximum Hurricane (PMH) and wind wave analysis.

We concur with the applicant's finding that the design water level should be 14.5 feet above the mean sea level datum for Units, Nos. 2, 3, 4 and 5. Although this value is acceptable, there are compensating errors in routing procedure employed.

If you have any further questions regarding this matter please let us know.

Sincerely yours,

Edward M. Willis

EDWARD M. WILLIS
Lieutenant Colonel, CE
Director

APPENDIX E

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

SEP 16 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by R. C. DeYoung is a review of the flood information presented in Amendment No. 19 to the Final Safety Analysis Report for Unit No. 2 Indian Point Nuclear Generating Station. It is presumed that the flood levels for all 3 units at the Indian Point Stations will be based on this amendment. Copies of our earlier reviews, for Unit No. 2 (Aug. 15, 1966) prepared by M. L. Meyer, and for Unit No. 3 (January 6, 1969) prepared by P. J. Carpenter, are attached.

This review was prepared by P. J. Carpenter and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

D. A. Nelson

Acting Director

Enclosures

Consolidated Edison Company of New York Inc.
Indian Point Nuclear Generating Station Unit No. 2
Docket No. 50-247

The probable maximum flood as defined by the U.S. Army Corps of Engineers, at the site, has been calculated as 1,[illegible],000 cubic feet per second. This discharge is approximately three times greater than the maximum observed flood at Green Island, and is approximately twice the maximum discharge observed for nearby [illegible]-sized drainage basins which appear to exhibit similar runoff characteristics. The stage for the maximum probable flood at the site, computed using standard step-backwater procedures, is given as varying between 13.4 and 14.0 ft msl (mean sea level) depending on concurrent tide levels at the Battery. It is shown that none of the dams on the Hudson River and its tributaries would fail during the probable maximum flood. The above results were obtained using conservative assumptions and appear to be reasonable.

The analyses show that the occurrence of the probable maximum flood on Esopus Creek would cause failure of Ashokan Dam some 75 miles upstream of the site. To establish a flood design level at Indian Point various combinations of the following factors were considered: 1) the flow resulting from the Ashokan Dam failure, 2) various concurrent Hudson River flood flows, and 3) various concurrent tide levels at the Battery. The results of these combinations of factors were compared with the stage of the probable maximum flood (14.0 ft msl) and the stage resulting from the probable maximum hurricane plus spring high tide (14.5 ft msl). The most critical combination investigated consisted of the flows from the Ashokan Dam failure caused by the probable maximum flood on Esopus Creek, the concurrent standard project flow (one half the probable maximum flood), the concurrent stage at the Battery corresponding to the standard project hurricane tide level and wind waves of one foot at the site. This stage is given as 15.0 ft msl. The lowest floor elevation of Unit No. 2 is given as 15.25 ft msl.

Other combinations of the above-mentioned factors, such as Ashokan Dam failure and the standard project hurricane or floods larger than the standard project flood on the Hudson River, could produce higher stages at the site. Depending on the degree of conservatism desired, any of these higher stages could also be selected as the design flood level. However, the stage for the combination selected for the design flood level exceeds those given for the probable maximum flood or probable maximum hurricane when these are considered as independent events.

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

APPENDIX F

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Consolidated Edison Company of New York, Inc.

Docket No. 50-247

by

N. M. Newmark
and
W. J. Hall

Urbana, Illinois

20 August 1970

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REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Indian Point Nuclear Generating Unit No. 2 for which application for a construction permit and an operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles N of the New York City boundary and 2.5 miles SW of Peeksill, New York.

This report is based on a review of the Final Facility Description and Safety Analysis Report (Ref. 1) and the containment design report (Ref. 2). The report also is based in part on the discussion and inspection resulting from the visit to the site on 2 May 1969 by N. M. Newmark and W. J. Hall in conjunction with Mr. K. Kniel and Mr. M. McCoy of AEC-DRL. A number of topics were discussed with the applicant and his consultants at the time of this visit, and subsequently additional information has become available through supplements to the FSAR and through discussions with the personnel of DRS, DRL, and the applicant and his consultants. A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of August 1966 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered there.

2

The design criteria for the containment system and Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration coupled with other appropriate loadings to provide for containment and safe shut down. The plant was also to be designed for an Operating Basis Earthquake of 0.1g maximum horizontal ground acceleration simultaneously with the other appropriate loads forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Dynamic Analyses

(a) Containment Building. The answer to Question 1.9 of the FSAR indicates that only the containment building, the primary auxiliary building, and the electric cable tunnel were designed with the use of semi-formal dynamic analyses. A description of the method of analysis employed is given briefly in Section 5.1.3.8 of the FSAR and in Section 3.1.5 of the containment design report. The procedure employed involved a calculation of the fundamental frequency and mode shape by use of a modified Rayleigh method. The base shear for the structure was computed from the period and the spectral response corresponding to the appropriate degree of damping. The base shear was then applied as a loading to the structure as an inverted triangular loading. The shears at the nodes were used to calculate the moments and displacements at various points in the structure. For the structures involved it is believed that the approach leads to a design which is reasonably adequate.

A similar approach was followed for the primary auxiliary building as described in the answer to Question 1.9. It is noted there that a one-third increase over working stress was allowed in the design of the bracing in the

case of the Design Basis Earthquake. This stress is below yield, and it is believed that the design will prove to be satisfactory.

(b) Other Buildings and Equipment. The discussion presented in answer to Question 1.9 of the FSAR for other buildings and equipment such as the control building, fan house, intake structure, etc., indicate that a refined static approach was used, which involves employing the peak value from the appropriate response spectrum curve for a given value of damping and multiplying this by the appropriate mass to obtain the inertial loading. From the description given for the various buildings and items of equipment, and the modeling techniques employed, it is concluded that the inertial loadings used in design are reasonably close to those that might be obtained with a more sophisticated analysis and lead to reasonable design values.

The submission in Question 1.3 of Supplement 13 indicates that the Turbine Building, and Fuel Storage Building Structure above the Fuel Storage Pit were reanalyzed by a multi-degree-of-freedom modal dynamic analysis method to check their adequacy. As a result of this reanalysis, the applicant advises that certain structural modifications will be made to columns and cross bracing in the Turbine Building to insure that it can withstand the DBE. The superstructure of the fuel storage building was ascertained to be adequately designed, without modification to withstand the effects of the DBE. The applicant states that reanalysis of the strengthened turbine building and superheater building for Indian Point No. 1 does not significantly affect the responses calculated for the original structures.

(c) Piping Analysis. The method used by the applicant for analysis of the piping, as described in the answer to Question 1.6 of the FSAR, is the same as was used in Ginna. The peak ground response spectrum value for 0.5 percent damping was used, applied as static accelerations in each direction

4

separately, and the resulting stresses superposed. It was assumed by the applicant that the piping was supported along rigid systems and therefore not subjected to amplified ground motion at points of support. The system was analyzed with the anchors and supports as actually used, according to the discussion presented to us during the time of our visit in May, 1969. It was the view of the applicant that the thermal motions were greater than any differential ground displacements and the latter therefore are not critical items in the design. In answer to Question 1.13 (Suppl. 13) the applicant advises that relative seismic displacement was considered for the main steam lines, where the largest relative displacements are expected; stress differentials of less than 10% resulted. Also, seismic supports installed to date are those specified in the design and employed in the analyses; where deviations in supports must occur, reanalysis will be carried out. These results and approaches appear satisfactory to us.

Since this plant was designed before recent developments and changes in piping design specifications, the 1968 ASME Addenda were not applied. Blow-down and earthquake were considered as separate items and not combined in this design. We are advised that the response to Question 1.9 of Supplement 12 states that a review of the Indian Point 3 reactor coolant system which is identical to Indian Point 2, for combined earthquake and blow-down indicates that the design is adequate.

It is stated in the answer to Question 1.6 of the FSAR that the approach resulted in a seismic design load approximately equal to 0.60W horizontally and 0.40W vertically taken simultaneously. It is further stated that for the Design Basis Earthquake the sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code

allowable stresses. In a similar manner the stresses in the pipe supports and hangers were limited to 1.2 times code allowable stresses.

The applicant originally made use of the maximum spectrum value only and no modal analyses were made; in other words only a static analysis with uniform accelerations was made. Consideration was not given to modified distribution of the inertial loading to take account of the combination of modal effects.

The response to Question 1.9 of Supplement 8, describing more detailed analyses of the reactor coolant system, feedwater lines, surge lines and typical steam lines by more formal methods as carried out later lends confirmation to the adequacy of the design. On this basis, there is reason to believe that the design is adequate.

Backfill Surrounding Containment Vessel

Nine feet of crushed rock backfill was placed between the external wall of the reinforced concrete containment vessel and the retaining wall holding back the rock on the uphill side. This crushed rock backfill is drained at the bottom to avoid water pressure against the containment structure. The fill is approximately 60 to 70 feet higher on one side of the structure than on the other because of the slope of the rock surface. The design, as discussed in Section 3.1.5 of the containment design report, considered local inertial forces of loose rock as an added loading against the containment pressure vessel, and also considered passive pressures caused by failure of the rock along the surface behind the retaining wall. The localized loadings from these forces were considered in the design of the containment structure and the discussion presented in the containment design report provides reasonable assurance that the containment vessel is capable of resisting these localized forces.

Class I Equipment in Structures other than Class I

The turbine building is Class III and not designed for earthquake loadings. The answer to Question 1.3 of the FSAR indicates that the only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the control building, main steam piping and feedwater piping, all of which could possibly be endangered by the Class III turbine building. It is further indicated there that no special provisions have been provided for protection except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the turbine building, it is not anticipated by the applicant that there will be any structural failure in this area. Our judgment as to the adequacy of this aspect of the design is based on the statement given in the application. And, in this respect, the answer to Question 1.3 (Supplement 13) which describes the analysis and strengthening of the Turbine Building and Superheater Building for Indian Point Unit No. 1, and their ability to withstand the DBE, should give additional protection for the control room.

It is further stated that the only Class III crane whose failure could endanger any Class I function is the fuel storage building crane and that the failure of this crane will not impair a safe and orderly shutdown. The answer to Question 1.3 (Suppl. 13) indicates that the only potential for crane lift off will be in the unloaded condition with the trolley parked over the support; the applicant advises that the unloaded crane will not be parked over the pool, so no hazard exists. It is also noted in the answer to Question 1.1.3 that the manipulator crane in the containment building,

a Class III crane, is restrained from overturning and will not endanger Class I structures.

Deformation Criteria

The general stress criteria applicable to the seismic design are summarized in Appendix A of the FSAR. The statement given on page A3 of Appendix A states that for all components, systems and structures classified as Class I, the primary steady state stresses, when combined with seismic stresses resulting from the response to the Design Basis Earthquake, are limited so that the function of the component system or structure shall not be impaired so as to prevent a safe and orderly shut-down of the plant.

We were advised at the time of our inspection of the plant in May 1969 that, for normal loadings plus the Operating Basis Earthquake, the intention was to use code allowables plus the 20 percent increase for transient conditions on Class I components and systems. For the Design Basis Earthquake and blow-down, basically the same criteria were used, although originally it had been planned to adopt higher allowables going into the plastic range using the code for faulted conditions. In actuality, as described in the answer to Question 1.7 of the FSAR, the allowable stresses in the case of the Design Basis Earthquake were limited to the yield point, or slightly below (see answer to Question 1.3 of Supplement 13).

The only references that we note where there was a calculation of stresses exceeding the yield point were at several places in the containment design report where it was mentioned that the calculations indicate that there could be possible local yielding of the liner under certain loading combinations, but that this would be limited and not be expected to be of a nature as to cause concern with regard to the integrity of the liner.

Reactor Internals

The mechanical design and evaluation of the reactor core and internals is described generally in Section 3.2.3 of the FSAR. From the discussion given it appears that the core support structure and core barrel have been designed with proper attention to support points and limitations of motions. The design criteria for the internals themselves, and specifically with reference to deflections under abnormal operation, are given in Table A.3-2 of the FSAR. These appear reasonable and should provide an adequate margin of safety.

Large Penetrations

A finite element analysis of the large penetrations in the containment vessel was made by the Franklin Institute and a description of the analysis and the results obtained is presented in the containment design report. Several analyses were made for different load combinations, and in addition a number of hand calculations were made to check the order of magnitude of the expected forces and stresses and to verify that the results were reasonable Our review of the material presented, to the extent possible, indicates that the penetration design is adequate.

Splices in Large Reinforcing of Bars

Cadweld splices were used in general in the construction of the containment vessel. We were advised that the early splices, about 10 percent of the total, were made with a bronze base, and the remaining 90 percent were made with ferritic base filler metal. Around the hatch opening, we observed that there was approximately a three foot stagger of adjacent splices, but in questioning we learned that there may not be such a stagger over other areas of the containment vessel. Lack of stagger of adjacent splices could

lead to planes of weakness and cause cracking under conditions of over-loading. The pressure tests, however, will reveal any such cracking.

Approximately one in 200 splices was removed for test purposes. This is generally adequate.

Instrumentation and Controls

At the time of the May 1969 visit it was ascertained that the applicant considers the control room as a Class I structure and intends that the housing of it will also be subject to Class I requirements. However, the instrumentation for the control room as well as other instrumentation critical to containment and safe shutdown, has been purchased from the vendors according to applicant's specifications. The answer to Question 1.9 describes the vibration tests employed for selected items of essential equipment; the purpose of these tests is to help demonstrate that little or no difficulty will be expected in the operating characteristics thereof under seismic conditions. Although not absolute proof of acceptability, satisfactory test results certainly help to confirm the adequacy of such instrumentation and control items. Further information on the design and procurement approach for protection system equipment is given in the answer to Question 7.27 (Suppl. 13), and lends confirmation to the approach adopted.

Tornado Loadings

The information contained in Section 3.4 of the containment design report, and the answer to Question 5.7 of the FSAR indicates that the structure is designed for the usual wind loadings. The analyses described in Appendix B of Supplement 6, indicate that the containment building can resist the design tornado. What effect if any that a tornado could have on the control room or other critical facilities is not stated. However, the applicant states that

the siding of the control room can resist wind velocities up to 162 mph, and the girts (supporting the panels) will fail at 0.62 psi negative pressure; the building is protected by other buildings on the south and west.

Steel Liner and Containment Vessel

The analyses that have been carried out with regard to the liner are summarized in the FSAR and some additional information is presented in the containment design report. It is our understanding that where bulges of the liners occurred during construction, of less than 2 in., nothing was done to correct the bulges. However, when bulges were 2 in. or greater the liner was pushed back into a position of not more than 2 in. away from its intended position, and additional studs were used to anchor the liner in place. Temporary bracing was employed to hold it in position until the concrete was cast. Because of the foregoing, and since the temperature rise in the lower part of the structure in the liner is reduced by the use of insulating material, it is not expected that the departures from the intended original surface will lead to any difficulties.

Proof Test Procedures and Instrumentation

It is our understanding that a detailed description of the proof test procedures is to be submitted at a later date. At the time of our visit in May 1969 it was proposed by the applicant that strain readings be taken only on the liner around the penetrations. We suggested that additional readings be made which would include diameter changes of the penetrations and other measurements that can be made conveniently and without excessive expense to provide evidence that the design meets the design criteria. Fig. 5.13-4 suggests that such readings will be made. In any event, an

interpretative report on the measurements that are taken should be provided and should be correlated with the calculations to provide evidence of validity of the design calculations.

Protection of Pipe Lines for Service Water

We were advised that pipelines for service water are embedded in the ground without any special protection. However, there appear to be alternate lines, although they are generally in the same location and/or trenches. In view of the foundation conditions surrounding the plant, and since there is no indication of previous fault motion or potential faulting, this design approach appears to be adequate. If redundancy in critical water supply is desired, it would be preferable to have separate water lines following independent routes.

Seismograph Installation

The answer to Question 1-1 of Supplement 3 indicates that one seismograph will be installed in the yard area, to provide further evidence of the extent of seismic excitation to which the plant might be subjected if an earthquake occurs. This is acceptable to us.

Containment Design Report

The containment design report, prepared for the applicant by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, has proven to be helpful in arriving at an evaluation of many of the factors inherent in the design. The tables presented are useful in helping to arrive at decisions as to the adequacy of the design; we commend those responsible for the preparation of this summary type material.

We should like to encourage this type of approach to studies of the containment, structures, piping, equipment and other Class I items. We should like to urge that attention be given also to summaries and tabulation of the most important information, in terms of stresses and deformations, including the sources of the various stress components, how they were combined, and related discussion and explanatory material (including figures) which would lend itself to a much better basis for judgment as to the adequacy of design of nuclear facilities in general.

CONCLUDING REMARKS

On the basis of the information made available to us concerning the Class I structures, piping, reactor internals, and other Class I items, it is our belief that the plant possesses a reasonable margin of safety to meet the original design requirements, including the imposed Design Basis Earthquake loading conditions.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Vols. I through V including Supplements 1, 2, 4, 5, 6, 7, 8 and 13," Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., AEC Docket No. 50-247, 1969 and 1970.

2. "Containment Design Report," for Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., prepared by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, March 1969. (Labeled Final Draft)

3. "Adequacy of the Structural Criteria for Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2," by N. M. Newmark and W. J. Hall, August 1966.

W. J. Hall

APPENDIX G

UNITED STATES
DEPARTMENT OF THE INTERIOR
OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

OCT 1 6 1970

Dear Mr. Chairman:

Pursuant to Section 5 of Public Law 89-605 as amended and other authorizations, we are presenting the views of the Department of the Interior in the matter of the application by the Consolidated Edison Company for an operating license for Indian Point Nuclear Generating Unit No. 2, Buchanan, New York, AEC Docket No. 50-247 (Amendment No. 9). The following comments incorporate those submitted by the Federal Water Quality Administration, the Fish and Wildlife Service and the Bureau of Outdoor Recreation.

The unit under review is the second of three units completed or being constructed at the Indian Point site. We note that applications for construction permits for two more units to be located approximately one mile south of the Indian Point site were made in June 1969.

The Department of the Interior does not object to the issuance of the operating license to the Consolidated Edison Company for Unit No. 2 of the Indian Point Nuclear Power Plant. Our position is based upon the firm commitment by the Company as expressed in its responses to the Atomic Energy Commission that it will meet the water quality standards applicable to the receiving waters and that it will take whatever steps are necessary to mitigate any harmful effects that operation of the plant may have on the fishery resources of the Hudson River and tributary waters.

The Company should be commended for the cooperation it has extended to representatives of this Department during the course of our review. The studies which the Consolidated Edison Company is presently engaged in indicate the Company's concern for the potential damages to the environment that could result from operation of this unit and the others planned at and in the vicinity of Indian Point.

We are pleased to note that the Company has made provisions to open part of its land holdings for compatible public recreation use. We express the hope that the Company's public use plans will be finalized and fully implemented at the earliest possible time.

Consolidated Edison has initiated or participated in a number of studies to determine the effects of both radiological and thermal discharges from the Indian Point reactors upon both the temperature distribution and the aquatic life of the Hudson River through its consultants, Quirk, Lawler and Matusky Engineers, and the Alden Research Laboratories of Worcester Polytechnic Institute. The Company has conducted mathematical studies of the probable temperature in the River and has checked these estimates with hydraulic model studies and actual field studies. In addition, Consolidated Edison has supported several independent but coordinated studies of the micro-organisms and aquatic life in the Hudson River and the probable effects of temperature and salinity changes upon them in the vicinity of the Indian Point Plant.

These studies are continuing and have been and will be helpful in assessing the effects of the Indian Point Unit No. 2 and of the other thermal plants which are proposed for construction on the shores of the Hudson River in the vicinity of Indian Point.

We have been provided information on plans for environmental monitoring of radiological and thermal releases proposed as a part of the operating license application. We understand that the plans for water quality monitoring, including radiological concentrations in the environment in microscopic and macroscopic aquatic life are acceptable to the State of New York. They appear reasonable and are considered generally acceptable to the Department of the Interior.

Through the monitoring programs the Company should have the necessary information to control its activities in a manner that will not violate applicable New York State as well as Federal water quality standards, recommendations of any enforcement conference or hearing board approved by the Secretary or order of any court under Section 10 of the Federal Water Pollution Control Act, and/or other State and Federal water pollution control regulations.

In view of the extensive and valuable fish and wildlife resources in the project area, it is imperative that every possible effort be made to safeguard these resources. Therefore, it is recommended that the Consolidated Edison Company be required to:

1. Continue to work closely with the Department of the Interior, New York State Department of Health, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys as planned. These surveys should include but not be limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the reactor effluent outfall.

   b. Beta and Gamma radioactivity analysis of selected plants and animals (including mollusks and crustaceans) collected as near the reactor effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and provide five copies to the Secretary of the Interior prior to project operation.

4. Conduct post-operational radiological surveys similar to that specified in recommendation (2) above, analyze the data, and prepare and submit reports every six months during reactor operation or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to appropriate State and Federal agencies for evaluation.

In addition to the above, the Atomic Energy Commission should urge the Consolidated Edison Company to:

1. Meet with the Department of the Interior, New York State Department of Environmental Conservation, New York State Department of Health, and other interested Federal and State agencies at frequent intervals to discuss new plans and evaluate results of the Company's ecological and engineering studies;

2. Conduct post-operational ecological surveys planned in cooperation with the above named agencies, analyze the data, prepare reports, and provide five copies of these reports to the Secretary of the Interior every six months or until the results indicate that no significant adverse conditions exist;

3. Construct, operate, and maintain fish protection facilities at the cooling water intake structure as needed to prevent significant losses of fish and other aquatic organisms; and

4. Modify project structures and operations including the addition of facilities for cooling discharge waters and reducing concentrations of harmful chemicals and other substances as may be determined necessary.

We appreciate the opportunity to provide these comments.

Sincerely yours,

Walter J. Hickel

Secretary of the Interior

Honorable Glenn T. Seaborg
Chairman, United States
Atomic Energy Commission
Washington, D. C. 20545

APPENDIX H
CONSOLIDATED EDISON COMPANY OF NEW YORK
DOCKET NO. 50-247
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1969	1968	1965
Long-term debt	$1,981.6	$1,901.6	$1,711.0
Utility plant (net)	3,793.3	3,583.6	3,169.5
Ratio - debt to fixed plant	.52	.53	.54
Utility plant (net)	3,793.3	3,583.6	3,169.5
Capitalization	3,818.4	3,667.6	3,228.1
Ratio - net plant to capitalization	.99	.98	.98
Stockholders' equity	1,836.7	1,766.0	1,517.1
Total assets	4,069.6	3,845.4	3,387.0
Proprietary ratio	.45	.46	.45
Earnings available to common equity	93.1	95.7	89.9
Common equity	1,210.2	1,139.0	1,072.1
Rate of return on common equity	7.7%	8.4%	8.4%
Net income	127.2	128.5	111.8
Stockholders' equity	1,836.7	1,766.0	1,517.1
Rate of return on stockholders' equity	6.9%	7.3%	7.4%
Net income before interest	198.0	193.9	168.4
Liabilities and capital	4,069.6	3,845.4	3,387.0
Rate of return on total investment	4.9%	5.0%	5.0%
Net income before interest	198.0	193.9	168.4
Interest on long-term debt	84.3	77.0	62.7
No. of times fixed charges earned	2.3	2.5	2.7
Net income	127.2	128.5	111.8
Total revenue	1,028.3	982.3	840.2
Net income ratio	.124	.131	.133
Operating expenses (incl. taxes)	830.5	788.3	668.6
Operating revenues	1,028.3	982.3	840.2
Operating ratio	.81	.80	.80
Retained earnings	426.1	400.9	321.7
Earnings per share of common	$2.47	$2.57	$2.42

	1969		1968	
Capitalization at 12/31	Amount	% of Total	Amount	% of Total
Long-term debt	$1,981.6	51.9%	$1,901.6	51.9%
Preferred stock	626.6	16.4	627.0	17.1
Common stock	1,210.2	31.7	1,139.0	31.0
	$3,818.4	100.0%	$3,667.6	100.0%

Moody's Bond Ratings:
First Mortgage Bonds — A

Dun and Bradstreet Credit Rating — AaA1

EXHIBIT K

November 16, 1970

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

BUCHANAN, NEW YORK

DOCKET NO. 50-247

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1.0 INTRODUCTION

The Consolidated Edison Company of New York, Inc., (applicant) filed with the Atomic Energy Commission (AEC or Commission) an application dated October 15, 1968, for an operating license for its Indian Point Nuclear Generating Unit No. 2. Indian Point Unit 2 has been under construction since issuance of a provisional construction permit on October 14, 1966.

Indian Point Unit 2 is located on a 227-acre site on the east bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York.

Indian Point Unit 2 is the first of the four-loop, current generation Westinghouse pressurized water reactor designs. It will be owned and operated by the Consolidated Edison Company of New York, Inc. The Westinghouse Electric Company (Westinghouse) is the principal contractor and has turnkey responsibility for the design, construction, testing, and initial startup of the facility. Westinghouse contracted with United Engineers and Constructors as architect engineer. Construction of the plant was performed by United Engineers until December 1969 when this function was assumed by WEDCO, a wholly-owned subsidiary of Westinghouse.

The operating license application is for a power level of 2758 megawatts thermal (MWt), the same as was requested in the construction permit application. Our evaluation of the engineered safety features

(with the exception of the emergency core cooling system) and our accident analyses, have been performed for a maximum power of 3216 MWt.

Our evaluation of the thermal, hydraulic, and nuclear characteristics of the reactor core and the performance of the emergency core cooling system was for a power rating of 2758 MWt. Before operation at any power level above 2758 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the core can be operated safely at the higher power level.

Our technical safety review of the design of this plant has been based on Amendment No. 9 to the application, the Final Facility Description and Safety Analysis Report (FFDSAR), and Amendments Nos. 10-25, inclusive. All of these documents are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D.C. The technical evaluation of the design of this plant was accomplished by the Division of Reactor Licensing with assistance from the Division of Reactor Standards and various consultants to the AEC.

In the course of our review of the application, many meetings were held with representatives of the applicant to discuss the plant design and proposed operation. As a consequence of our review, additional information was requested, which the applicant provided by amendments to the application. A chronology of the principal actions relating

to the processing of the application is attached as Appendix A to this safety evaluation. In addition to our review the Advisory Committee on Reactor Safeguards (ACRS) independently reviewed the application and met with both the AEC staff and the applicant on several occassions to discuss the plant. The ACRS report on Indian Point Unit 2, dated September 23, 1970, is attached to this Safety Evaluation as Appendix B. Appendices C through G include reports from our consultants on meteorology, hydrology, seismic and structural design, and radiological monitoring. Appendix H contains the staffs evaluation of the applicant's financial qualifications.

Based upon our evaluation of the plant as summarized in subsequent sections of this report, we have concluded that Indian Point Unit 2 can be operated at thermal power levels of up to 2758 MWt without endangering the health and safety of the public. Subsequent to the issuance of an operating license the unit will be required to operate in accordance with the terms of the operating license and the Commission's regulations under the surveillance of the Commission's regulatory staff.

2.0 FACILITY DESCRIPTION

Indian Point Unit 2 is one of three reactors currently planned for the Indian Point site. Indian Point Unit 2 is adjacent to Indian Point Unit 1, a 615 MWt pressurized water reactor plant that has been in operation since August 1962. Indian Point Unit 3, a plant similar to Indian Point Unit 2, received a provisional construction permit in August 1969, and is presently under construction at the Indian Point site. Each unit has its own auxiliary systems and safety features. The three units, however, will share a common inlet water canal and a common discharge canal. In addition, the controls for Indian Point Unit 2 and Indian Point Unit 1 are located in separate portions of a common control room.

The Indian Point Unit 2 pressurized water reactor is fueled with slightly enriched uranium dioxide in the form of ceramic pellets contained in zircalloy fuel tubes. Water serves as both the moderator and the coolant. Heat is removed from the reactor core by four separate coolant loops, each provided with a separate pump and steam generator. The heated water flows through the steam generators where heat is transferred to the secondary (steam) system. The water then flows back to the pumps to repeat the cycle. The system pressure is controlled by the use of a pressurizer in which steam and water are maintained in thermal equilibrium.

The secondary steam produced in the steam generators is used to drive the turbine generator. The heat of condensing steam is rejected to the circulating water system and discharged to the Hudson River. The condensate is then recharged to the steam generators to repeat the secondary cycle.

The primary coolant system includes the reactor, steam generators, primary coolant pumps, primary coolant piping, and the pressurizer. This system is housed inside the containment building which is a steel-lined, leak-tight reinforced concrete structure. The containment provides a barrier to the release to the environment of radioactive fission products that might be released inside the containment in the event of an accident. Auxiliary systems, including the chemical and volume control systems, the waste handling system, and additional auxiliary cooling systems, are housed separately, principally in the adjacent primary auxiliary building. The primary auxiliary building also houses components of the engineered safety features. A separate fuel handling building is provided for storage of spent fuel. A separate turbine building houses the turbine generator.

Control of the reactor is achieved by reactivity control using top entry control elements that are moved vertically within the core by individual control drives. Boric acid dissolved in the coolant is used as a neutron absorber to provide long-term reactivity control.

To assure reactor operation within established limits, a reactor protection system is provided that automatically initiates appropriate actions whenever plant conditions monitored by the system approach preestablished limits. The reactor protection system acts to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The engineered safety features include an emergency core cooling system that will cool the reactor core in the event of an accident that results in loss of the normal coolant, containment cooling and iodine removal systems that provide for removal of heat and radioactive iodine from the containment atmosphere should such action be required, and a hydrogen control system that will limit the accumulation of hydrogen within the containment in the event of a loss-of-coolant accident. A containment penetration pressurization system and seal water injection system are provided to assist in isolating the containment in the event of an accident and prevent the escape of fission products to the environment outside the plant.

3.0 SITE AND ENVIRONMENT

3.1 Population and Land Use

The Indian Point site consists of 227 acres in the town of Buchanan in upper Westchester County, New York, approximately 24 miles north of the New York City boundary line. The estimated population distribution in the vicinity of the site is presented in table 2.1.

TABLE 2.1

CUMULATIVE POPULATION

Distance (miles)	1960 (U.S. Census)	1980 (Projected)
0-1	1,080	2,100
0-2	10,810	20,900
0-3	29,630	59,520
0-4	38,730	78,800
0-5	53,040	108,060
0-10	155,510	312,640

The minimum radius of the exclusion area* for Indian Point Unit 2 is 520 meters. The applicant has chosen 1100 meters as the outer

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*Exclusion area is defined in the Commission's Site Criteria, 10 CFR Part 100, as that area surrounding the reactor in which the reactor licensee has the authority to determine all activities including removal of personnel and property from the area.

boundary of the low population zone** because of the limited population within this distance from the plant.

The Commission's site criteria guidelines state that the population center distance*** should be at least 1-1/3 times the distance from the reactor to the outer boundary of the low population zone (LPZ), but also state that in applying this guide due consideration should be given to the population distribution within the population center. The nearest corporate boundary of Peekskill (population 19,000) is approximately 800 meters (0.5 miles) from Indian Point Unit 2. Because of the limited population within the low population zone (66) including that portion of Peekskill within the zone, and because Peekskill is of a generally industrial nature in the vicinity of the site and the resident population within and out to 1-1/3 times the low population zone distance is low, we concluded at the time of our construction permit review that the distance selected by the applicant for the exclusion area radius, the LPZ outer boundary, and the population center distance meet the intent of the 10 CFR Part 100 guidelines and are acceptable. On the basis of our evaluation of the potential radiological consequences of postulated design basis accidents,

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**Low population zone is defined in the Commission's Site Criteria, 10 CFR Part 100, as the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.

***Population center distance is defined in the Commissions Site Criteria, 10 CFR Part 100, as the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

we conclude that the calculated doses presented in Section 11.0 of this evaluation are well within the guidelines of 10 CFR Part 100 for these distances.

3.2 Meteorology

The meteorology of the Indian Point site is affected by its position in a deep river valley. Consequently, the wind direction generally follows a pronounced diurnal cycle with unstable flow in the up-river direction during the daytime and stable flow in the down-river direction at night.

The applicant has presented the results of meteorological measurements taken at the site over a period of two years including windspeed, wind direction, and temperature lapse rate data for various heights. We have reviewed the data presented and conclude that they provide an adequate basis for selecting the meteorological parameters used in determining the routine effluent release limits and in evaluating the consequences of postulated accidents. The comments of our meteorological consultants, the Environmental Science Service Administration (ESSA) support this conclusion and are attached as Appendix C.

3.3 Geology and Seismology

During our review of this site prior to issuance of the construction permit for Indian Point Unit 2, we and our consultant, the U. S. Geological Survey, concluded that the geology of the site provides an adequate founding medium for the plant buildings and

structures. No new developments have occurred during the construction permit review of Indian Point Unit 3 or otherwise since our construction permit review for Indian Point Unit 2 to change our previous conclusion on the acceptability of the geological and seismological features of the Indian Point site.

Maximum ground accelerations of 0.10g and 0.15g were used for the Operating Basis Earthquake* and the Design Basis Earthquake**, respectively. These values were selected at the time of the construction permit review. At that time we and our consultant, the U. S. Coast and Geodetic Survey, concluded that they were acceptable for the site.

A strong motion seismograph has been installed on a concrete slab directly on bedrock in the yard area of the plant to record data related to ground motion in the event of a seismic disturbance at or near the site. These data would be employed in an evaluation of the effects of the seismic disturbance to assure the capability for continued safe operation of the plant.

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*"Operating Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems and components, necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

**"Design Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems, and components, necessary to shut down the reactor and maintain the unit in a safe shutdown condition without undue risk to the health and safety of the public are designed to remain functional.

3.4 Hydrology

The applicant has reevaluated the potential flooding that could occur at the site. The following hypothetical flood conditions were analyzed: (1) the probable maximum flood peak discharge of 1,100,000 cubic feet per second resulting from the probable maximum precipitation occurring over the total basin, a 12,650 square mile area above the plant site; (2) the flooding caused by failure of the Ashoken Dam concurrent with a major river basin flood (standard project flood) with a peak discharge of 705,000 cubic feet per second and a hurricane storm surge (standard project hurricane), and (3) the probable maximum hurricane concurrent with the high spring tide in the Hudson River. These three hypothetical floods are the most severe of several investigated, and each of the three results in a maximum water surface elevation of about 15 feet above mean sea level. We have reviewed the method of calculation and conditions assumed and find that they are conservative and acceptable. Both the U. S. Geological Survey and the Coastal Engineering Research Center provided consulting services with respect to our flooding evaluation. Their reports are attached as Appendix D and Appendix E, respectively.

3.5 Environmental Monitoring

The radioactivity levels in the vicinity of the Indian Point site have been measured by the applicant since 1958 to ascertain the

impact of operation of Indian Point Unit 1 on the background levels of radioactivity. The environs of the Indian Point site have been studied intensively for many years by the Institute of Environmental Medicine at New York University Medical Center. These studies concerned both the exposure to man and to the flora and fauna indigenous to the Hudson River. All the results compiled to date indicate that radioactive effluents from Indian Point Unit 1 operation have produced barely quantifiable radiation exposure to the public and have had no detectable effect on the ecology of the area.

The operational environmental radiation monitoring program for Indian Point Unit 2 will be a continuation of the present program. The program includes direct measurements of gamma radiation and analyses to monitor fallout, air particulates, airborne iodines, water from various surface drinking water supplies, Hudson River water, water from lakes near the site, well water, lake aquatic vegetation, Hudson River vegetation, river bottom sediments, river aquatic biota, terrestrial vegetation, and soil. The report of the U. S. Department of the Interior is attached as Appendix G. This report incorporates the comments of the Federal Water Quality Administration, the Fish and Wildlife Service, and the Bureau of Outdoor Recreation. The report comments favorably on current activities being performed by or for the applicant in connection with determining the effects

of both radiological and thermal discharges at the plant site. Recommendations for continued effort in the area of environmental monitoring and ecological studies are included in the report. This report has been forwarded to the applicant.

We conclude that the applicant's program will be adequate for monitoring the radiological effects of Indian Point Unit 2 operations on the environment and for assessing the effects of releases of radioactivity to the environment from operation of the plant on the health and safety of the public.

4.0 REACTOR DESIGN

4.1 General

The nuclear reactor for Indian Point Unit 2 was designed and manufactured by Westinghouse. The principal design features, materials of construction, and arrangement of various components of the Indian Point Unit 2 core are the same as those for the Rochester Gas and Electric Company's R. E. Ginna facility (Docket No. 50-244), which has been licensed for operation by the Commission and which has completed almost a full year of power operation. Further, the zircalloy clad fuel, burnable poison in the initial core loading, a chemical neutron absorber, and part-length control rods to shape axial power distribution are used in substantially the same manner in both the Ginna and the Indian Point Unit 2 reactors. On the basis of our previous review of all of these features for the Ginna reactor, we conclude that these same features are acceptable for Indian Point Unit 2.

4.2 Nuclear Design

The Indian Point Unit 2 reactor core differs principally from the Ginna and Connecticut Yankee (Docket No. 50-213) reactor cores in that the Indian Point Unit 2 reactor core is somewhat larger. The Indian Point Unit 2 core is about 23% greater in cross sectional area and 20% longer than the Connecticut Yankee core and about 89% greater in cross sectional area and the same length as the Ginna core. Because this larger core could be subject to power

oscillations or power tilts, we reviewed the nuclear design and power distribution detection and control systems for the Indian Point Unit 2 reactor core in detail.

During plant operation, changes in the core power level or the control rod configuration can cause time-dependent variations in the local power distribution as a result of variations in the concentration of fission products and their radioactive decay products. The most significant fission product-decay product chain with regard to core behavior is the decay of iodine-135 to xenon-135 since the latter is a strong absorber of thermal neutrons. The local oscillations in the neutron flux and in the power level can occur even though the average power level of the core is maintained constant, and the magnitude of the oscillations may decrease, remain constant, or increase with time.

The spatial stability of the xenon distribution and resultant core power peaking abnormalities for the Indian Point Unit 2 core have been investigated by Westinghouse with the conclusion that the core is stable against various types of xenon induced spatial oscillations in the X,Y horizontal plane. This conclusion is supported by analysis and by experiments performed in the Connecticut Yankee reactor. An initial test program for Indian Point Unit 2 will be performed to verify this stability. If this initial test program does not demonstrate stability, the applicant has agreed to operate with partially inserted control rods, or to add fixed or burnable poison shims sufficient to assure stability

through reduction of the moderator temperature coefficient, or to operate at reduced power levels. Because of the test program that will be performed and the operating limitations that will be imposed if required, we conclude that the reactor will be stable with respect to potential power oscillations in the X,Y horizontal plane.

The analysis made by Westinghouse indicates that the reactor may be subject to divergent xenon oscillations in the axial direction, resulting in an axial power distribution imbalance or tilt. In view of this, it is assumed that the axial power tilts can occur, and provision is made to detect and control differences in the fraction of the total power generated in the upper and lower halves of the core. Data correlations have been made at the Connecticut Yankee reactor and at the Ginna reactor to relate the readings obtained from the split out-of-core detectors to axial power tilts. Additional correlations will be established during the Indian Point Unit 2 startup tests. Part-length control rods are provided to prevent unacceptable axial power tilts and to control potentially divergent axial xenon spatial oscillations. Analytical studies and experience with the Ginna reactor, provide assurance that any axial oscillations can be controlled such that the power distribution will be maintained within design limits. In addition, automatic protective action is provided to avoid exceeding design power peaking factors at full power in the event of control system malfunctions. To accomplish this, the overtemperature ΔT and overpower ΔT trip set points are automatically reduced in proportion to the axial

power tilt as measured by the split out-of-core neutron detectors. We conclude that the system of detection instrumentation, control with part length rods, and automatic protection for potential axial power tilts is acceptable.

Even in the absence of xenon induced instability, power tilts or imbalances can occur in the horizontal or axial planes as a result of control rod misalignment. Analyses for Indian Point Unit 2 and experiments in the Connecticut Yankee reactor have shown that these power tilts can be detected by (1) the split out-of-core neutron detectors, (2) the core exit thermocouples, or (3) the movable in-core neutron detectors. All of these detectors are required to be operable by the Technical Specifications. In addition detection will ordinarily be readily accomplished by the fixed in-core neutron instrumentation.

The power distribution in the Indian Point Unit 2 core is expected to be stable or only slowly varying within known limits and adequate core instrumentation will always be available to detect, monitor, and diagnose any significant power maldistributions.

We conclude that the Indian Point Unit 2 reactor core nuclear design and instrumentation is acceptable.

4.2 Thermal-Hydraulic Design

We have evaluated the adequacy of the core thermal and hydraulic design, both for steady-state plant operation and for anticipated plant transients. The design criteria selected by the applicant to prevent fuel damage are: (1) the departure from nucleate

boiling (DNB) ratio (determined using the Westinghouse W-3 correlation) shall not be less than 1.3 during normal plant operation or as a result of anticipated transients; and (2) no fuel melting shall occur during either normal operation or anticipated transient conditions. The anticipated plant transients that result in the most severe core thermal transients are loss of coolant flow, excessive load increase, and a loss of external electrical load. The applicant's analyses show that the DNB ratio will be greater than 1.3 for each of these plant transients when operating at the license power level of 2758 MWt. The lowest DNB ratio calculated as a result of any of the plant transients, was for the case of simultaneous loss of electrical power to the four reactor coolant pumps. This transient results in a DNB ratio of 1.42. In addition, no fuel melting is predicted to occur for steady-state operation or as a result of anticipated transients.

As stated above the Indian Point Unit 2 reactor core is designed to undergo anticipated plant transients with a minimum DNB ratio greater than 1.3. On this basis, clad temperature should not be significantly affected by a transient and no fuel failure should occur for the range of fuel element burnup planned for the Indian Point Unit 2 core. As part of a continuing experimental effort to

demonstrate satisfactory performance of fuel at high burnup and high power density, Westinghouse is continuing a fuel irradiation program at conditions significantly in excess of current PWR design limits, and will establish power burnup limits for the fuel. These irradiation programs are being conducted at both the Saxton and Zorita reactors. Sustained operation of selected fuel rods at peak design power levels in the Zorita reactor will increase assurance that the fuel has adequate margins to accommodate transient overpower operation.

Based on our evaluation of the results of these analyses, and on our review of the design limits and the operating experience of similar reactors, we conclude that the reactor core thermal and hydraulic design is acceptable for operation at the rated power of 2758 MWt.

5.0 REACTOR COOLANT SYSTEM

5.1 General

The reactor primary coolant system, including all vessels, pumps, and piping is designed for a pressure of 2485 psig and a temperature of 650°F. The system has been designed to withstand, within the stress limits of the codes used in the design, the normal loads of mechanical, hydraulic, and thermal origin, plus those due to anticipated transients and the operating basis earthquake.

5.2 Primary System Components

The reactor internals are designed to withstand the normal design loads of mechanical, hydraulic, and thermal origin, including those resulting from anticipated plant transients and the operating basis earthquake, within the stress limit criteria of Article 4 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. Although the Indian Point Unit 2 reactor internals are not designed to withstand simultaneously the loads resulting from loss-of-coolant accident blowdown and seismic events, the applicant has submitted a summary of an analytical study of the behavior of the reactor internals under simultaneous blowdown and seismic loadings (WCAP-7332-L). The results of this study indicate that for the combined blowdown and design basis earthquake loadings the resulting deflections are within the loss-of-function limits except for the control rod immediately adjacent to the coolant line that was assumed to fail. On the

basis that the core reactor internals remain functional and that adequate shut down margin can be achieved by control rod insertion, we conclude that the stress and deflection limits for the combined blowdown and design basis earthquake loadings provide an adequate margin of safety.

The primary system side of the steam generators, the pressurizer, and the main coolant pump casings, have been designed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition - Summer 1969 Addenda, as Class A vessels. For other Class I pumps, valves, and heat exchangers the inspection program required independent review of (1) the physical and chemical test data for pressure boundary materials, (2) radiographs of valve bodies, valve bonnets and pump casings, and (3) dye-penetrant examinations of heat exchanger tubes and welds. These requirements resulted in fabrication and inspection programs that contain the essential elements of the recently proposed ASME Codes for Nuclear Pumps and Valves. We find the design codes and inspection requirements acceptable.

We have reviewed the information submitted by the applicant with respect to operating limitations on heatup and cooldown of the primary system imposed by the fracture toughness properties of the materials of the Indian Point Unit 2 reactor vessel. Our evaluation was based on a proposed redraft of section NB-2300 Special Materials Testing (Section III ASME Boiler and Pressure

Vessel Code) dated July 28, 1970, which reflects the material testing requirements in a form consistent with the AEC Fracture Toughness Criteria. As a consequence of our evaluation the applicant has agreed to the heatup and cooldown limitation as presented in Section 3.1-B of the Technical Specifications which represents a modification of his initial submittal. On the basis that these limits reflect a very conservative method of defining pressure vessel fracture toughness, we conclude that they are acceptable.

5.3 Coolant Piping

The reactor coolant piping has been designed in accordance with the requirements of the American National Standards Institute (ANSI) B31.1 Code for Power Piping, 1955 Edition, including the requirements of Nuclear Code Cases N-7 and N-10. All welding procedures and operators were qualified to the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. Additional inspection requirements for the reactor coolant piping during fabrication included ultrasonic and dye-penetrant inspection of all pipe welds. Non-destructive examination of valves included radiographic examination of the valve castings and ultrasonic inspection of all forged components. Dye-penetrant surface examination was also performed. With this program, the inspection of the Indian Point Unit 2 reactor coolant piping substantially

meets the requirements of Class 1 systems under the ANSI B31.7 Code for Nuclear Power Piping adopted in 1969. On this basis we have concluded that the design and inspection program for this system is acceptable.

The original seismic design analysis for the Indian Point Unit 2 reactor coolant system utilized only static methods of analysis. Recently, at our request, the applicant completed a rigorous dynamic analysis of this system utilizing both modal-response spectra and model time-history methods of analyses. As with the reactor internals, the combined loading of a concurrent loss-of-coolant accident blowdown and design basis earthquake was not considered in the design of the Indian Point Unit 2 reactor coolant system. However, the applicant recently completed an analysis of the response of the reactor coolant system to be installed in Indian Point Unit 3 for these combined loads. Since the Indian Point Unit 3 and the Indian Point Unit 2 reactor coolant systems are identical, the applicant has used the results of the analysis for Indian Point Unit 3 in conjunction with the material properties for the Indian Point Unit 2 piping, as determined from tests, to determine that the combined seismic and accident loads can be tolerated by the Indian Point Unit 2 reactor coolant system within acceptable stress limits.

Based on our review of the design limits and analytical procedures employed, we find that the design of the Indian Point Unit 2 reactor coolant system is acceptable.

5.4 Other Class I* (Seismic) Piping

At our request the applicant performed additional seismic analysis on other Class I piping. The adequacy of the seismic design of the feedwater lines, pressurizer surge line, and a typical steam line has been confirmed by a dynamic analysis utilizing the modal-response-spectra method. The adequacy of the seismic design of other Class I (Seismic) piping in the plant was determined by performing a dynamic analysis on selected "worst case" systems. Several systems that are the most vulnerable to dynamic excitation because of system flexibility or location in the supporting structure were analyzed and the resulting stresses compared with the stresses determined by the original static analyses. The applicant has concluded that the conservatism of the original static analysis provided adequate margins to accommodate the previously undetermined dynamic effects.

Based on our review of the original static methods employed and the confirmatory evidence obtained from the recent dynamic analyses of the most vulnerable systems, we have concluded that the design of the Class I (Seismic) piping systems in Indian Point Unit 2 is acceptable.

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*See Section 6.1 for definition of Class I structures, systems, and components.

5.5 Inservice Inspection

An inservice inspection program for the reactor coolant system is included in the Technical Specifications. This program follows Section XI of the ASME Code, Rules for Inservice Inspection of the Reactor Coolant System, as closely as practical. The design of the primary system including the capability to remove insulation at selected areas provides an acceptable degree of access for inspection purposes. The applicant also intends to conduct periodic inservice inspections of the primary pump motor flywheels.

The applicant will review the inservice inspection program with us after five years of reactor operation. It may then be modified based on experience gained during these five years. At that time, we will also require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity.

5.6 Missile Protection

We have reviewed the applicant's primary system layout within the containment in terms of the protection afforded the containment liner and Class I (seismic) systems inside the containment from missiles that might be generated as a result of a primary system failure. We have concluded that adequate protection from potential missiles is provided by the system arrangement and surrounding thick circumferential concrete walls and the concrete floors.

The primary pump motor flywheels installed in Indian Point Unit 2 are the same as those in use in other plants. The flywheels are the standard Westinghouse design, fabricated of A 533B steel. On the basis of the use of high grade material, extensive quality control measures, special manufacturing procedures and preservice and inservice surveillance requirements, we have concluded that assurance has been provided that the integrity of the flywheels will be maintained.

5.7 Leak Detection

The reactor coolant pressure boundary leak detection systems for this plant are similar to those we have reviewed and found acceptable for other plants using a Westinghouse nuclear steam supply system. The systems are based upon air particulate monitoring, radiogas monitoring, humidity detection, and containment sump level monitoring. These systems provide an array of instrumentation that is sensitive, redundant, and diverse and that has adequate alarm features. The sensitivity of these systems is consistent with their primary purpose of detecting any leak in the primary system pressure boundary which could be indicative of incipient failure. The Technical Specifications require that two reactor coolant leak detection systems of different principles shall be in operation when the reactor is operated at power. We conclude that the leak detection systems for Indian Point Unit 2 are acceptable.

5.8 Fuel Failure Detection

The fuel element failure detection system will measure delayed neutron activity in one hot leg of the reactor coolant system. The monitor is connected in series with a delay coil to allow a decay time for N^{16} gamma activity (half life of 7.1 seconds) of about 60 seconds before the coolant reaches the detector. This delay reduces gamma ray background and facilitates detector sensitivity. An alarm signal is provided for the channel. We conclude that this system which is inherently faster in response than previous systems reviewed for other reactors is acceptable.

5.9 Vibration Monitoring and Loose Parts Detection

The major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. Vibration analyses for the core support barrel considered inlet flow impingement and turbulent flow. Natural frequency calculations were made to assure that there would be no deleterious response to known excitations such as pump blade passing and driven frequencies. Fuel bundle response to anticipated driving forces has been calculated and determined by tests in the Westinghouse Reactor Evaluation Center.

The vibration monitoring system to be used for the preoperational test program on Indian Point Unit 2 will consist of mechanical gauges to measure gross relative motion between the thermal shield and core barrel, strain gauges on selected guide tubes, and

accelerometers on the upper core plate. We have concluded that the vibration design analyses and the preoperational test program are acceptable.

In the course of our review of the Indian Point Unit 2 application, it has been noted that techniques for the analysis of neutron noise spectra and accelerometer measurements on the lower heads of primary system vessels might be developed to provide a useful method for inservice monitoring of reactor coolant systems to detect changes in the vibration of reactor components or the presence of loose parts. The applicant has stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefullness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

5.10 Conclusion

Based on our review of (1) the codes and standards used for design, (2) the fabrication and inspection procedures, (3) the inservice inspection program, (4) the provisions for missile protection and leak detection, (5) the provision for fuel failure detection, and (6) the provisions for preoperational vibration

testing and the developmental effort for inservice monitoring to detect vibrations and loose parts, we have concluded that the design and inspection procedures for the reactor coolant system for the Indian Point Unit 2 are acceptable.

6.0 CONTAINMENT AND CLASS I (SEISMIC) STRUCTURES

6.1 General Structural Design

The applicant has categorized as Class I (seismic) those structures (e.g., containment structure and primary auxiliary building), and those systems and components (e.g., reactor vessel and internals, emergency core cooling system), whose failure could cause a significant release of radioactivity or that are vital to the safe shutdown of the facility and the removal of decay heat. We have reviewed the applicant's classification of structures, systems, and components and conclude that they have been classified appropriately.

The Class I (seismic) structures at Indian Point Unit 2 are the containment structure, the primary auxiliary building, the control room building, the fuel storage pool, the diesel generator building, and the intake structure and service water screenwell. The major portion of the primary auxiliary building, the fuel storage pool, and the intake structure are of reinforced concrete construction. The control room building, the diesel generator building, the fuel storage building and the non-Class I portions of the primary auxiliary building are constructed of steel framing with composite metal panel siding.

The environmental conditions that were considered in the structural design include the operating basis earthquake (OBE), the design basis earthquake (DBE), the flooding and wind due to

the probable maximum hurricane, and the flooding due to the probable maximum flood. We have concluded that these conditions were used for the design in an acceptable manner.

6.2 Structural Design and Analysis

The Indian Point Unit 2 primary containment has a free volume of 2.6×10^6 cubic feet and a design pressure of 47 psig. The containment structure is a right cylinder (thickness 4.5 ft) with hemispherical dome (thickness 3.5 ft) mounted on a flat (thickness 9 ft) base mat. The reinforced concrete is lined with 1/4 inch minimum thickness welded ASTM A442 grade 60 firebox quality carbon steel plate. The reinforcing bars conform to ASTM A432 specifications. The reinforcing in the cylinder wall is placed in horizontal and vertical directions with added diagonal tangential reinforcing for earthquake resistance. The reinforcing bars conform to ASTM A432 specifications. Cadweld splices are used in 14S and 18S bars.

We have evaluated the pressure transients that might occur in the containment in the event of a loss-of-coolant accident assuming various sizes of primary coolant system breaks. For the range of postulated break sizes up to and including the double-ended severance of the largest reactor coolant pipe, the largest calculated peak containment pressure is 40 psig. The design pressure of the containment exceeds the calculated peak pressure by more than 10% and is acceptable.

The containment is designed to remain within the elastic range for the 0.10g OBE concurrent with the accident and other applicable loads. It is also designed to withstand the 0.15g DBE concurrent with the accident without loss of function.

We and our seismic design consultant, Nathan M. Newmark, are in agreement with the loading combinations and allowable stresses used by the applicant. Stress and strain limits conform to the requirements of ACI 318-63, Part IV-B. The ACI load factors have been replaced by factors suitable for concrete containment structures.

Based on our review of the design of the containment structure and its capability to withstand the predicted pressures from potential accidents, we conclude that the structural design aspects of the containment are acceptable.

In evaluating the capability of the Class I (seismic) structures, systems, and components, to withstand the dynamic loads due to seismic events, our seismic design consultant, Nathan M. Newmark Consultant Engineering Services, considered the geology and nature of the bedrock, design loads and load combinations, the seismic design parameters, and methods of analysis. On the basis of our review and that of our seismic design consultant, we conclude that the Class I (seismic) structures, systems, and components of Indian Point Unit 2 are designed to accommodate all applicable loads and are acceptable. The report of our seismic design consultant is attached as Appendix G.

During our review we noted a limited number of cases where failure of non-Class I (seismic) structures could potentially endanger Class I (seismic) structures and equipment. These included the Indian Point Unit 1 superheater stack and superheater building, the turbine building, and the fuel storage building. In response to our concern, the applicant performed analyses of these structures using a multi-degree of freedom modal dynamic analysis method, to determine the modifications needed to assure that gross structural collapse of these structures would not occur in the event of a DBE. As a result of these analyses, additional seismic reinforcement is being provided for both the superheater building and the turbine building and the Indian Point Unit 1 superheater stack is to be reduced in height by 80 feet. The truncation of the stack is to be accomplished at a convenient time in the next three years and prior to operation of Indian Point Unit 3. We and our seismic design consultant have reviewed the material submitted by the applicant and conclude that the dynamic analyses performed, and the design modifications proposed, are acceptable.

We have reviewed the as-built wind resistance of Class I structures at the Indian Point Unit 2 facility. Analysis indicates that both the containment and reinforced concrete portions of the primary auxiliary building and intake structure can sustain winds in the range of 300 miles per hour. The control building and diesel generator building which are constructed of structural steel with composite metal panel siding, are estimated by the applicant to be capable of sustaining wind loads of up to 160 miles per hour.

is being installed on the containment for strength testing, although examinations will be made for cracking and distortion during the pressure test. Periodic leakage rate tests will be performed on the containment and its penetrations.

We have concluded that the provisions for testing and surveillance of the containment are acceptable. Test and surveillance requirements are included in the Technical Specifications.

6.4 Missile Protection

The possibility exists that missiles might be generated in the unlikely event of a failure of the turbine generator. Although the design criteria for Indian Point Unit 2 did not include consideration of protection against missiles resulting from turbine failures, at our request the applicant has assessed the protection available against missiles that might result from a turbine failure at the maximum overspeed condition (133% of rated normal speed). Specific provisions have been added to limit the off-site consequences that could result from a missile failure, and to provide for safe shut down of the unit. These include an alternative cooling water supply for the charging pumps and added missile protection for a potentially vulnerable portion of the auxiliary steam generator feedwater lines. In addition, a second completely independent turbine speed control system has been provided to reduce the probability of a runaway speed condition that might result in a turbine failure. This

system is designed to the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Criteria No. 279 for protection systems. The Technical Specifications require periodic testing of the overspeed devices to assure operability. We conclude that the applicant has made appropriate provisions to reduce the probability of a destructive turbine missile from being generated and affecting Class I (seismic) items.

The Indian Point Unit 2 reactor vessel cavity is designed to protect the containment against missiles that might be produced by postulated failure of the reactor vessel. Failure of the reactor vessel would result in fluid jet-reaction forces in the cavity wall adjacent to the vessel split or crack as well as stress in the cavity wall from a rise in cavity pressure, both of which would result from coolant blowdown. Also reaction forces in the cavity wall and floor might be produced by the impact of missiles generated by pressure vessel failure. By the use of extensive steel reinforcing, the concrete cavity has been designed to resist both fluid jet and missile impact forces that could result from pressure vessel failure by either longitudinal splitting or various modes of circumferential cracking. The cavity is also designed to sustain a fluid pressure rise to 1000 pounds per square

inch. We have reviewed the applicant's analysis and conclude that the cavity as designed provides adequate protection for the containment liner against missiles that might result from a postulated pressure vessel failure.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling System

The principal equipment of the emergency core cooling system consists of (1) three 50% capacity high pressure safety injection pumps, (2) two 100% capacity residual heat removal pumps for low pressure injection and external recirculation, (3) two 100% capacity recirculation pumps for recirculation internal to the containment, (4) one 100% capacity boron injection tank, and (5) four 33-1/3% capacity accumulators. This system provides redundant capability to inject borated cooling water rapidly into the core in the event of a loss-of-coolant accident and to maintain coolant above the level of the core for an indefinite period following the accident.

The applicant's evaluation of the performance of these systems is based on detailed analyses of (1) the hydraulic behavior of the primary coolant system during and subsequent to a loss-of-coolant accident, and (2) the thermal response of the core during the same period. The analytical methods used to predict the hydraulic behavior of the primary coolant system during a loss-of-coolant accident have been improved significantly during the construction period for Indian Point Unit 2. The original analysis presented in Volume 4 of the FFDSAR was performed with the FLASH-1 hydraulics computer program. This program is limited to a three-node

We conclude that the emergency core cooling system will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad water reaction to less than 1% of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for cooling is lost and before the clad is so embrittled as to fail upon quenching and (4) reduce the core temperature and then maintain core and coolant temperature levels in a subcooled condition until accident recovery operations can be accomplished.

In summary, we conclude that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident.

The emergency core cooling system design as presently installed at Indian Point Unit 2 was reviewed by the Division of Reactor Licensing during 1967, subsequent to the issuance of the construction permit on October 14, 1966. This system represented a complete redesign, a considerable increase in flow capability, and enhanced performance when compared to the system reviewed for the construction permit. On the basis that the very significantly improved performance of the redesigned emergency core cooling system provides additional assurance for limiting clad temperatures and maintaining a coolable core we concurred with the applicant's decision to remove the reactor pit crucible from the facility design.

7.2 Containment Spray and Cooling Systems

Two independent heat removal systems are provided to control the containment pressure and temperature following a loss-of-coolant accident. Each system, acting alone at its rated capacity, will prevent over-pressurization of the containment structure. The two systems are the containment spray system and the fan cooling system. The design of each is substantially the same as the design of systems provided at the Ginna plant and other licensed plants.

The containment spray system consists of two 50% capacity spray pumps and is sized to limit the containment post-accident pressure to below design pressure. Sodium hydroxide and boric acid are used as additives to the spray solution to remove radioactive iodine which might be present in the containment after an accident. We have reviewed the use of these chemical spray additives in terms of their iodine removal capabilities, and in addition have evaluated the chemical compatibility of the spray solution with other reactor components. As a result of our review, we conclude that the spray system is adequately sized to cool the containment, that the alkaline spray solution will reduce the iodine concentration in the containment atmosphere, and that corrosion of other materials used in the containment does not introduce a safety problem.

The containment fan cooling system provides complete redundancy to the containment spray system for heat removal from the containment atmosphere during post-accident conditions. Five 20% capacity fan

coolers are provided. Since the fan coolers are located within containment, they must be capable of operating in the post-accident environment. Westinghouse has conducted an environmental test program to demonstrate this capability. Our evaluation of these tests, including the heat removal capability of the heat exchangers, and environmental and radiation testing of the fan cooler motors, valve motor operators and electric cabling indicates that these components will function satisfactorily in the accident environment. An iodine-impregnated charcoal filter system has been included with the fan cooler system to remove organic iodine from the post loss-of-coolant containment atmosphere. The charcoal beds are preceded by demisters and high efficiency particulate air (HEPA) filters.

We have evaluated the inorganic and organic iodine removal capability of the charcoal beds on the basis of tests with steam - air mixtures at 100% relative humidity following prolonged flooding of the bed. We conclude that inorganic and organic iodine removal efficiencies of 90% and 10% per pass, respectively, are conservative values that are justified by the available information.

In summary, we have reviewed the containment spray and fan cooling systems in terms of (1) capability to control the containment temperature, (2) capability to remove inorganic and organic iodine,

(3) system and component redundancy, and (4) capability to function in the post-accident containment environment. We conclude that there is reasonable assurance that these systems will operate as proposed subsequent to a loss-of-coolant accident.

7.3 Containment Isolation Systems

In addition to the usual capability of isolating all lines leading to and from the containment, the Indian Point Unit 2 containment is provided with additional systems to minimize the potential leakage of fission products subsequent to an accident. A containment penetration and weld-channel pressurization system provides for continuous pressurization of zones enclosing containment penetrations and the welds in the containment liner. The system continuously maintains an overpressure of clean, dry air that is in excess of the containment design pressure. Pressurized zones include each piping penetration, each electrical penetration, double gasketed spaces on the personnel and equipment hatches, and the channels over weld seams of the containment liner. The air pressure is maintained by the instrument air compressors with backup from the plant air compressors and from a standby source of nitrogen cylinders. Pressure indication and alarm instrumentation is provided locally and in the control room to assure that loss of pressure will be detected and corrected.

In addition, an isolation seal water system has been provided to assure containment isolation by (1) injecting seal water between the seats and stem packing of the globe and double disc isolation valves used on larger lines, and (2) injecting seal water directly into the line between the closed diaphragm valves used in the smaller lines penetrating containment. Seal water injection is provided for all lines connected to the reactor coolant system and for lines that may be exposed to the containment atmosphere subsequent to an accident. Although the use of the seal water system following a loss-of-coolant accident provides an additional means of reducing leakage, we have not considered the effect of this system in determining the offsite radiological consequences.

We have concluded that the capability provided for isolating the containment is acceptable.

7.4 Post-Accident Hydrogen Control System

In the event of a loss-of-coolant accident, radiation from the core and from escaped fission products will dissociate some of the cooling water into gaseous hydrogen and oxygen. Continued evolution of hydrogen would increase the concentration in the containment to a point where ignition could occur and thus provide an additional energy source.

Redundant flame recombiner units are installed within the Indian Point Unit 2 containment. Each unit has the design capability to prevent the ambient containment hydrogen concentration from exceeding two percent by volume. The units are designed to function, following the loss-of-coolant accident in a containment pressure environment of 1 to 5 psig. Each recombiner system consists of (1) a flame recombiner unit located within containment, (2) a control panel located outside of containment, and (3) a hydrogen gas stand located outside of containment. On the basis of (1) our detailed review of the design of the system and its controls, (2) satisfactory performance testing of the device, and (3) satisfactory environmental testing of those portions of the recombiner system installed within the containment, we conclude that there is reasonable assurance that the recombiner system will perform its intended post-accident function.

In addition, the applicant will provide the capability for purging the containment atmosphere through appropriate filters as an alternate backup means of hydrogen control. The containment penetrations to be used for this system are installed. The design and installation of the equipment required will be performed during the first two years of operation at power.

8.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

8.1 Reactor Protection and Control System

The reactor protection system instrumentation for Indian Point Unit 2 is the same as that installed at the Ginna plant. The adequacy of the protection system instrumentation was evaluated by comparison with the Commission's proposed general design criteria published on July 11, 1967, and the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 2 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the primary loops out of service the reactor is normally automatically limited to 60% of full power. However by manual adjustment of several protection system set points in a manner consistent with the Technical Specifications adequate reactor protection can be provided for operation up to 75% of full power.

We have reviewed the applicant's analysis of the seismic response of the protection system instrumentation and associated electrical equipment and find that adequate testing has been performed on the nuclear instrumentation, switch gear, and process system instrumentation.

In connection with our review of potential common mode failures we have recently considered the need for means of preventing common failure modes from negating scram action and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has been responsive to our request for information and has provided the results of analyses which indicate that the consequences of such transients are tolerable for the existing Indian Point Unit 2 design at a power level of 2758 MWt. Although additional study is required of this general question, we conclude that it is acceptable for the Indian Point Unit 2 reactor to operate at a power level of 2758 MWt while final resolution of this matter is made on a reasonable time scale.

8.2 Initiation and Control of Engineered Safety Features

The instrumentation for initiation and control of engineered safety features for the Indian Point Unit 2 is the same as that installed at the Ginna plant. This basic design has been reviewed extensively in the past and we consider it to be acceptable.

We have reviewed the capability for testing engineered safety feature circuits during reactor operation. Resistance tests will be used for routine determinations of the operability of the master and slave relay coils. The circuits upstream of these relays can be partially tested during operation. During plant shutdown, circuits can be tested completely by coincident tripping of instrument channels and a consequent operation of the master and slave relays in the entire downstream initiating system. We have concluded that this

testing capability is acceptable for Indian Point Unit 2.

8.3 Off-Site Power

Two 138 kilovolt (kV) lines connect the Buchanan switchyard to the Millwood switching station, which in turn is connected to the Consolidated Edison grid and the Niagara Mohawk and Connecticut Light and Power systems. Two additional 138 kV lines, using a separate route from the first two lines, connect the switchyard to the Orange and Rockland tie.

The applicant stated that an analysis of the transmission system has indicated that the system is stable for the loss of any generating unit including Indian Point Unit 2.

A single 138 kV line connects the Buchanan switchyard to Indian Point Unit 2. In addition, three 13 kV lines connect the switchyard to Indian Point Unit 1. Three 138/13 kV transformers in the switchyard feed these three 13 kV lines. While the 138 kV system is the normal supply for the auxiliary load associated with plant engineered safety features, one of the three Indian Point Unit 1 13 kV lines is available to provide power via automatic switching to Indian Point Unit 2 through a 13/6.9 kV transformer. By switching circuit breakers in Indian Point Unit 1, the other two 13 kV lines can also be made available to provide power to Indian Point Unit 2. As the 13/6.9 kV supply is not capable of carrying the total plant auxiliary load for Indian Point Unit 2, the main coolant pumps and the circulating water pumps must be tripped off before the supplies are switched.

We conclude that the off-site power supply provides an adequate source of power for the engineered safety features and safe shutdown loads.

8.4 Onsite Power

Onsite power is supplied by three independent diesel generator sets connected in a separate bus configuration such that there is no automatic closure of tie breakers between the three buses to which the generators are connected. The redundant engineered safety feature (ESF) loads are arranged on the three separate buses such that failure of a single bus will not prevent the required ESF performance under accident conditions. The design engineered safety feature and safe shutdown loads per diesel generator are 1813, 2210, and 2353 HP for the first one-half hour following a loss-of-coolant accident. The loads are then changed to 2438, 2235, and 2043 HP for the recirculation phase of the emergency core cooling system operation. On the basis of our evaluation, we have determined that the appropriate diesel generator ratings are 2200 HP continuous, and 2460 HP for 2,000 hours. We note that some of the estimated emergency loads are above the continuous rating of the machines, but below the 2,000 hour ratings. We consider that this margin is acceptable for Indian Point Unit 2.

Each diesel generator is started automatically upon initiation of emergency core cooling system operation or upon under-voltage on its corresponding 480-volt emergency bus. The generators are

housed in a separate Class 1 (seismic) structure. On-site diesel fuel storage capacity provides a minimum of seven days operation at the required safety feature loads. These design and operating features are acceptable for Indian Point Unit 2.

Our review of the ac auxiliary power system has disclosed that there is adequate capacity and an adequate degree of physical and electrical separation of redundant features. The 125 volt dc system consists of two individually housed batteries. The dc system is divided into two buses with a battery and battery charger for each bus. Each of the two station batteries has been sized to carry its expected loads for a period of two hours following a plant trip at a loss of all ac power.

We conclude that the onsite emergency power system is acceptable.

8.5 Cable Installation

We have reviewed the applicant's cable installation relative to the preservation of the independence of redundant channels by means of separation, and relative to the prevention of cable fires through proper cable rating and tray loading. This has been performed by reviewing the cable installation criteria and method of layout design and by field inspection of electrical cable installation during construction.

A single electrical tunnel carries the electrical cables from the electrical penetration area of the containment to the control building. This tunnel carries all of the electrical cables except the power cables for the reactor coolant pumps, the pressurizer

heater cables, and the control rod power cables. The cables in the tunnel are arrayed on either side of a three-foot aisle in trays or ladders. Separation is provided for in the form of distance, metal separators, or transite barriers. The electrical tunnel does not contain any spliced cable connections. Therefore, the probability of a fire is reduced. Further, a fire detection system and an automatically operated water spray system are provided in the tunnel. Tunnel cooling is provided for by redundant cooling fans. On the basis of adequate separation within the tunnel, a minimum number of heat producing cables and features, redundant cooling systems, and fire detection and spray systems we conclude that the single electrical tunnel is acceptable.

Sixty electrical penetrations are provided in a single electrical penetration area to provide for entry of signal, control, and power cables into the containment. The penetrations are located on three-foot centers, both horizontally and vertically, and are of the hermetically sealed type. As a result of our review, fire barriers in the form of transite sheets were added to separate the power cable penetration from the instrument and control cable penetrations. In addition, as a result of our review certain modifications were made to the cabling in the penetration area, including shortening of cable runs and elimination of cable loops. The segregation of power cables and the shortening of the cable runs reduces the probability of failure by fire and on this basis, we consider the single electrical penetration area acceptable for Indian Point Unit 2.

The applicant has performed a design audit to verify the separation of redundant engineered safety feature power and control electrical cabling. A design review of instrument cabling was also performed on a sample basis.

On the basis of our review of cable installation at Indian Point Unit 2, we conclude that the resulting cable layout, as installed, is acceptable.

8.6 Environmental Testing

Westinghouse has conducted an environmental test program for the instrumentation and controls that are located inside containment and that must function in the environment following a loss-of-coolant accident. We have reviewed the results of this testing program and conclude that the essential instrumentation and controls will function properly in the accident environment.

9.0 RADIOACTIVE WASTE CONTROL

Liquid and gaseous waste handling facilities are designed to process waste fluids generated by the plant so that discharge of liquid and gaseous effluents to the environment will be minimized. Liquid waste is processed both by direct removal of radioactive material with ion exchange resins and by evaporative separation. Using these methods the volume of radioactive waste will be greatly concentrated and the purified liquid streams will either be reused or discharged. Small quantities of radioactive liquid waste will be released routinely to the condenser circulating water discharge canal common to all three units where the waste will be diluted and discharged to the Hudson River.

The limits on routine radwaste releases from the three units that are planned for operation at the Indian Point site will require that the combined releases from the three units when added together be within the limits specified in 10 CFR Part 20. This requirement is stated in Section 3.9 of the Technical Specifications for both liquid and gaseous effluents.

The liquid effluent releases from the three nuclear facilities will be discharged from a common discharge canal into the Hudson River. The nearest sources of public drinking water supplies from the Hudson River are located at Chelsea, New York (backup water supply for New York City) and at the Castle Point Veterans Hospital, 22 and 20.5 miles upstream of the Indian Point site, respectively.

of the Commission's regulations. For gaseous halogens and particulates with half-lives greater than eight days, the applicable limits of the Technical Specifications are less than 1% of the limits given in 10 CFR Part 20. The Technical Specifications also require that the maximum release rate of gaseous waste not exceed the annual average limit.

Based on our review we conclude that the means provided by the applicant for the disposal of radioactive waste are substantially the same as those we have approved for other facilities and are acceptable. We also conclude that acceptable means are provided and will be used to keep the release of radioactivity from the plant within ranges that we consider to be as low as practicable.

10.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown include (1) the chemical and volume control system, (2) the residual heat removal system, (3) the component cooling system, and (4) the service water system. The systems necessary to assure adequate cooling for spent fuel include (1) the spent fuel pool cooling system, (2) the fuel handling system, and (3) the service water system. The designs for these systems are substantially the same as those we reviewed and found acceptable for the Ginna plant.

10.1 Chemical and Volume Control System

The chemical and volume control system (1) adjusts the concentration of boric acid for reactivity control, (2) maintains the proper reactor coolant inventory and water quality for corrosion control, and (3) provides the required seal water flow to the reactor coolant pumps. The amount of boric acid to be added to the core for reactivity control is determined by the operator. The addition of unborated water as a result of operator error could result in an unintentional dilution during refueling, reactor startup, and power operation. The applicant's analysis indicated that because of the slow rate of dilution there is ample time for the operator to become aware of the dilution and to take corrective action. The applicant is actively participating in the development of a device for continuous monitoring of the reactor coolant boron concentration and will evaluate the feasibility of installing such a monitor when developed.

Our review of the chemical and volume control system emphasized those portions involved in routine and emergency injection of concentrated boric acid. We conclude that the design is acceptable.

10.2 Auxiliary Cooling Systems

Subsystems for auxiliary cooling are the component cooling system, the residual heat removal loop, the spent fuel pool cooling loop, and the service water system. The piping for these three systems is designed to the ANSI B31.1 Code for Pressure Piping.

These systems are equivalent in purpose and design to those of other recently licensed plants. On the basis of our review of this plant and others using the similar systems, we have concluded that these systems are acceptable.

10.3 Spent Fuel Storage

The fuel handling system is designed to transfer spent fuel to the storage pool and to provide storage for new fuel. The spent fuel storage facility is basically the same in capacity and design as those used in previously licensed pressurized water reactor plants. The fuel pool is sized to accommodate spent fuel from 1-1/3 core loadings.

As in other designs, mechanical stops will be incorporated in the crane to restrict motion of the spent fuel cask to its assigned area, adjacent to one side of the fuel storage pool. In addition, the spent fuel racks in the area adjacent to the fuel cask storage

location would be used only in the event that a complete core is unloaded and one-third of a core from a previous unloading is already in storage.

The pool floor is located below grade level and founded on solid rock. Structural damage from a dropped fuel cask would not result in a rapid loss of water from the pool. Makeup water can be supplied from the demineralizer water supply at a flow rate of 150 gpm. Additional water can be provided in an emergency by the use of temporary hookups to other sources.

As a consequence of our evaluation of the potential consequences of a postulated fuel handling accident, the applicant has agreed to provide charcoal filters in the refueling building to reduce the calculated offsite doses that might result in the event of a fuel handling accident in the refueling building. The installation of the filters will be completed during the first year of full power operation.

We conclude that the designs of the spent fuel storage pool and the fuel handling system are acceptable.

11.0 ANALYSES OF RADIOLOGICAL CONSEQUENCES FROM DESIGN BASIS ACCIDENTS

11.1 General

In order to assess the safety margins of the plant design, a number of operating transients were considered by the applicant, including rod withdrawal during startup and at power, moderator dilution, loss of coolant flow, loss of electrical load, and loss of ac power. The reactor control and protection system is designed so that corrective action is taken automatically to cope with any of these transients. Based on our evaluation of the information submitted by the applicant and our evaluations of other PWR designs at the operating license stage, we conclude that the Indian Point Unit No. 2 control and protection system design is such that these transients can be terminated without damage to the core or to the reactor coolant boundary, and with no offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including a control rod ejection accident, an accident involving rupture of a gas decay tank, a steamline break accident, a steam generator tube rupture accident, a loss-of-coolant accident, and a refueling accident.

The calculated offsite radiological doses that might result from the control rod ejection accident, and the accident involving rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

The consequences of the steamline break and the steam generator tube rupture accidents can be controlled by limiting the permissible concentrations of radioactivity in the primary and secondary coolant systems. The Technical Specifications for the Indian Point Unit No. 2 facility limit the primary and secondary coolant activity concentrations such that the potential 2-hour doses at the exclusion radius that we calculate for these accidents do not exceed 1.5 Rem to the thyroid or 0.5 Rem to the whole body.

Our evaluations of the loss-of-coolant accident and the refueling accident are discussed in the following sections.

11.2 Loss-of-Coolant Accident

The design basis loss of coolant accident (LOCA) for the Indian Point Unit No. 2 plant is similar to that evaluated for other PWR plants in that a double-ended break in the largest pipe of the reactor coolant system is assumed.

Although the basis for the design of the emergency core cooling system is to limit fission product release from the fuel, in our conservative calculation of the consequences of the LOCA we have assumed that the accident results in the release of the following percentages of the total core fission product inventory from the core: 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens that are released from the core is assumed to plate out onto internal surfaces of the containment

building or onto internal components and is not available for leakage. We assume that 10% of the iodine available for leakage from the containment is in the form of organic iodide, and that 5% is in the form of particulate iodine. The reactor is assumed to have been operating at a power of 3217 MWt prior to the accident. The primary containment is assumed to leak at a constant rate of 0.1 percent of the containment volume per day for the first day and 0.05 percent per day thereafter. We evaluated the iodine removal capability of the sodium hydroxide containment spray system and assumed an inorganic iodine removal constant of 4.5 per hour for the spray system. We evaluated the iodine removal capability of the iodine impregnated charcoal filter system and assumed a removal constant of 0.49 per hour for inorganic iodine and a removal constant of 0.048 per hour for organic iodine. Iodine particulates are assumed to be removed by the high efficiency particulate air filters. The inhalation rate of a person offsite is assumed to be 3.5×10^{-4} cubic meters per second.

For the calculation of the two-hour dose at the site boundary we used an atmospheric dispersion factor corresponding to Pasquill Type "F" stability, with a 1 meter per second wind speed and an appropriate building wake effect. We calculated the potential doses at the site boundary for this 2 hour period to be 180 Rem to the thyroid and 4 Rem to the whole body. At the low population zone boundary our calculated potential doses for a 30-day period are 270 Rem to the thyroid and 7 Rem to the whole body.

In evaluating the above doses, no credit was given for the isolation valve seal water injection system, the penetration pressurization system, or the weld channel pressurization system. Operation of these systems, which interpose a high gas pressure or seal water area between the containment and the outside atmosphere at all points where leakage might occur, should significantly reduce the leakage rate from the containment, and, thus reduce the doses following an accident. These systems are well designed and tested, and should be available in the event of an accident (see Section 7.3). We did not consider the effect of these systems in our dose calculations because it is inherently difficult to accurately measure leakage rates of less than 0.1% per day by current testing methods.

The control room for Indian Point Unit No. 2 was not designed to meet the requirements we have imposed in more recent construction permit reviews, that the dose for the course of the accident to occupants of the control room be limited to 5 Rem to the whole body and 30 Rem to the thyroid. In order to provide additional protection to the control room occupants in the event of a loss-of-coolant accident, the applicant has equipped the control room with protective clothing and self-contained air respirators for the operators. In view of these provisions, we have concluded that the control room, as constructed, is acceptable in this regard.

11.3 Fuel Handling Accident

We have evaluated the potential consequences of a fuel handling accident, in which it is postulated that a fuel assembly is dropped in the spent fuel pool or transfer canal. We assumed that: (1) all 204 rods in the dropped bundle are damaged, (2) the accident occurs 90 hours after shutdown of the core from which the dropped bundle has been removed, (3) 20% of the noble gases and 10% of the iodine in the dropped fuel bundle are released to the refueling water and the dropped fuel bundle has been removed from a region of the core which has been generating 1.43 times the average core power, (4) 90% of the released iodine is retained in the refueling water, (5) the fission products released from the pool are discharged to the atmosphere by the building recirculation system through charcoal filters with an iodine removal efficiency of 90%, and (6) the same meteorological conditions exist as were assumed for the loss-of-coolant accident. The resultant calculated doses at the site boundary are 146 Rem to the thyroid and less than 4 Rem to the whole body.

11.4 Conclusions

We have calculated offsite doses for the design basis accidents that have the greatest potential for offsite consequences using assumptions consistent with those we have used in previous safety reviews of PWR plants and have found the resulting calculated doses to be less than the guideline values of 10 CFR Part 100.

(3) review and report upon all proposed changes to the Technical Specifications; (4) conduct unannounced spot inspections of plant monitoring operations; (5) review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plant; and (6) convene, at the request of the nuclear power generation manager or a nuclear plant general superintendent or chairman or vice chairman of the committee, to review and act upon any matter they may deem necessary.

Westinghouse will participate in the startup and initial operation of the plant and will continue to make available technical support to the Indian Point Unit 2 staff during operation of the facility.

We conclude that the applicant's organization is acceptably staffed and technically qualified to perform its operational duties subject to satisfactory completion of licensing examinations of personnel requiring licenses.

12.3 Emergency Planning

The site emergency plan for the Indian Point site describes the emergency organization and its responsibilities. The scope of the emergency plan includes consideration of local contingencies, site contingencies, general (off-site) contingencies, implementation levels for each contingency, notification channels, the support provided by civil authorities, protective measures for each

contingency, communications facilities, and training drills. The applicant has provided an extensive description of the medical support that will be available although it is not incorporated explicitly in the plan. The planned medical support provides for emergency treatment of plant personnel both at the site and at a designated hospital where facilities equipment and medical personnel to handle radiation contaminated injured personnel will be available.

We conclude that the applicant's emergency plan is acceptable for Indian Point Unit 2.

12.4 Industrial Security

The immediate plant area (restricted area), including Indian Point Unit 1 will be enclosed by a fence. Access to the restricted area for all personnel will be through manned gatehouses or locked gates which are under the direct control of the station security forces. Security guards will make routine patrols of all property within the site boundary and outside the restricted area and are required to make hourly reports to the central control room.

The controlled area of Indian Point Unit 2 will include the containment, the fuel storage building, the primary auxiliary building, and the emergency diesel generator building. Normal access to these areas is through the existing security room for Indian Point Unit 1. All other doors and hatches leading into the controlled area will be locked and will be supervised by means of door switches connected to the open door alarm board in the

security room, and the category alarm board in the Indian Point Unit 1 central control room. The containment personnel hatch doors have remote indicating lights and annunciators that are located in the control room and that indicate the door operational status.

Offsite applicant employees must identify themselves at the main gate prior to admission to the restricted area, receive approval for entry by the general superintendent or his designated representative, and sign in on an admission sheet. If access into the controlled area is approved, they must be accompanied by a qualified guide.

We conclude that the applicant has taken reasonable measures to provide for the security of the facility.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define safety limits and limiting safety system settings, limiting conditions for operation, periodic surveillance requirements, certain design features, and administrative controls for the operating plant. These specifications cannot be changed without prior approval of the AEC. The applicant's initial proposed Technical Specifications, presented in Amendment No. 20, have been modified as a result of our review to describe more definitively the allowable conditions for plant operation. The Technical Specifications as approved by the regulatory staff, may be examined in the Commission's Public Document Room.

Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and that means are provided for keeping the release of radioactivity from the plant within ranges that we consider as low as practicable. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features to mitigate the consequences of unlikely accidents will be available.

14.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS reported on the application for construction of the Indian Point Unit 2 at the proposed site in a letter dated August 16, 1966. The applicant has been responsive to the recommendations made by the ACRS in that letter, and we conclude that the matters raised have been resolved satisfactorily during the design and construction of the Indian Point Unit 2.

The ACRS reported on its review of the application for an operating license for Indian Point Unit 2 in their letter, dated September 23, 1970, attached as Appendix B.

In its letter, the ACRS made several recommendations and noted several items all of which have been considered in the indicated sections of our evaluation. These include: (1) reevaluation of potential flooding at the Indian Point site (Section 3.4), (2) additional seismic reinforcing at the Indian Point Unit No. 1 superheater building and truncation of the superheater stack (Section 6.2), (3) reactor design, power distribution, and control of potential xenon oscillations (Section 4.2), (4) containment design and isolation (Sections 6.2 and 7.3), (5) containment cooling and iodine removal systems (Section 7.2), (6) emergency core cooling system and removal of the reactor pit crucible (Section 7.1), (7) post-accident hydrogen control (Section 7.4),

(8) charcoal filters in the refueling building (Section 10.3), (9) reactor core instrumentation (Section 4.2), (10) reactor protection with only three of four loops in service (Section 8.1), (11) inservice vibration monitoring and loose parts detection (Section 5.9), (12) fuel failure detection (Section 5.9), (13) availability requirements for primary coolant leak detection systems (Section 5.7), (14) pressure vessel fracture toughness (Section 5.2), (15) integrity of high burnup fuel during design transients (Section 4.3), and (16) common mode failure and anticipated transients without reactor scram (Section 8.1).

The ACRS concluded in its letter that if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

15.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and all of the directors and principal officers of the applicant are United States citizens.

The applicant is not owned, dominated or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes, is involved. For these reasons and in the absence of any information to the contrary, we have found that the activity to be performed will not be inimical to the common defense and security.

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations that relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR Part 50.33(f) and 10 CFR Part 50 Appendix C. The Consolidated Edison Company's application as amended by Amendment No. 21 thereto, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated operating cost for each of the first five years of operation plus the estimated cost of permanent shutdown and maintenance of the facility in a safe condition. The estimated operating costs are $10.0 million for 1971 (the first year of operation), $14.8 million for 1972, $12 million for 1973, $10.9 million for 1974 and $10.7 million for 1975 (Amendment No. 21). Such costs include the costs of operating and maintenance and fuel. The applicant's estimate of the cost of permanently shutting down the facility and maintaining it in a safe condition is (1) $265,000 for the first year of shutdown and $50,000 for each year thereafter if the reactor core is removed from the vessel, and (2) $240,000 per year if the core is not removed.

We have examined the certified financial statements of the Consolidated Edison Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1969 financial report indicates that operating revenues

for 1969 totaled $1,028.3 million; operating expenses (including taxes) was $830.5 million; the interest on the long-term debt was earned 2.3 times; and the net income for the year was $127.2 million, of which $102.1 million was distributed as dividends to the stockholders, and the remainder of $25.1 million was retained for use in the business. As of December 31, 1969, Company's assets totaled $4,069.6 million, most of which was invested in utility plant ($3,793.3 million), and earnings reinvested in the business were $426.1 million. Financial ratios computed from the 1969 statements indicate a sound financial condition, (e.g., long-term debt to total capitalization--0.52, and to net utility plant--0.52; net plant to capitalization--0.994; the operating ratio--0.81; and the rates of return on common--7.7%; on stockholder's investment--6.9%; and on total investment--4.9%). The record of the Company's operations over the past 5 years reflects that operating revenues increased from $840 million in 1965 to $1,028 million in 1969; net income increased from $111.8 million to $127. million; and net investment in utility plant from $3,170 million to $3,793 million. Moody's Investors Service (August 1969 edition) rates the Company's first mortgage bonds as A (high-medium grade). The Company's current Dun and Bradstreet rating (July 1970) is AaAl.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR Part 50.33(f) with respect to the operation of Indian Point Unit 2. A copy of the staff's financial analysis is attached as Appendix H.

17.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities such as power reactors under 10 CFR Part 50.

17.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The Consolidated Edison Company, is with respect to Indian Point Unit 2, subject to the foregoing requirements, and has taken the following steps with respect thereto.

The Company has furnished to the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) Nos. NF-100.

Further, the Company executed Indemnity Agreement No. B-19 with the Commission as of January 12, 1962, which was amended to cover its pertinent preoperational fuel storage under license SNM-1108 on March 4, 1969. The Company has paid the annual indemnity fee applicable to preoperational fuel storage.

17.2 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from, preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million.

Accordingly, no license authorizing operation of Indian Point Unit 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended and that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. The amount of financial protection required for a reactor having the rated capacity of this facility would be $82 million. Consolidated Edison Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licensees, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

18.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the Consolidated Edison Company of New York, Inc., dated December 6, 1965, as amended (Amendments Nos. 9 through 25, dated October 15, 1968, October 13, 1969, October 24, 1969, November 21, 1969, December 29, 1969, January 27, 1970, March 2, 1970, March 30, 1970, April 17, 1970, June 3, 1970, July 14, 1970, July 17, 1970, July 28, 1970, July 29, 1970, August 13, 1970, August 28, 1970, and November 12, 1970, respectively) complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. Construction of the Indian Point Nuclear Generating Unit No. 2 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-21, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
5. The applicant is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The applicable provisions of 10 CFR Part 140 have been satisfied; and
7. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to Consolidated Edison for Indian Point Unit No. 2, the Commission's Division of Compliance will prepare and submit a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformaty of this construction to the provisional construction permit and the application. Before an operating license will be issued to Consolidated Edison for Indian Point Unit No. 2, assuming such a license is authorized following the public hearing, the facility must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to license issuance.

APPENDIX A

CHRONOLOGY OF

REGULATORY REVIEW OF THE CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2

(SUBSEQUENT TO CONSTRUCTION PERMIT NO. CPPR-21

ISSUED ON OCTOBER 14, 1966)

1.	April 17, 1967	Submittal of Amendment No. 6 containing design information on the Emergency Core Cooling System and other areas as requested by the ACRS in their letter to the Chairman AEC, of 8/16/66.
2.	July 18, 1967	Meeting with applicant to discuss revised design of Emergency Core Cooling System and other areas as per Amendment No. 6.
3.	August 2, 1967	Letter to applicant requesting additional information on subjects addressed by the ACRS in their letter of 8/16/66.
4.	October 16, 1967	Submittal of Amendment No. 7 in response to DRL request of August 2, 1967.
5.	October 31, 1967	Submittal of Amendment No. 8, revised pages for Amendment No. 7.
6.	December 28, 1967	ACRS Subcommittee meeting to discuss emergency core cooling system, reactor pit crucible, primary coolant system, other areas.
7.	January 30, 1968	Submittal of "Report on the Containment Building Liner Plate Buckle in the Vicinity of the Fuel Transfer Canal".
8.	February 2, 1968	Meeting with applicant to discuss content of Amendments No. 6, 7, and 8.
9.	February 13, 1968	Meeting with applicant to complete discussion of February 2, 1968.

10.	March 8, 1968	ACRS Full Committee meeting to discuss Emergency Core Cooling System; reactor internals; primary coolant system, design, fabrication, in-service inspection, and leak detection; core design; reactor pit crucible; and containment liner quality control and stress analysis.
11.	October 15, 1968	Consolidated Edison Company filed application for an Operating License for the IP-2 Plant. Amendment 9, Volumes 1, 2, 3, & 4.
12.	March 5, 1969	AEC-DRL requested additional information on medical and emergency plans.
13.	March 12, 1969	AEC-DRL staff met with Con Ed personnel to discuss scheduling of regulatory review of application for operating license.
14.	April 3, 1969	AEC-DRL staff met with Con Ed personnel to discuss structural and seismic design and tornado protection.
15.	April 16, 1969	AEC-DRL staff met with Con Ed to discuss accidental and normal radioactivity release from the IP-2 plant.
16.	April 28, 1969	Con Ed requested extension of completion date for construction of the IP-2 plant.
17.	May 2, 1969	AEC-DRL staff and Nathan M. Newmark, seismic design consultant, met with Con Ed personnel at the IP-2 site to discuss seismic design and review status of construction and site inspection.
18.	May 19, 1968	AEC-DRL staff issued an order extending completion date for construction of the IP-2 plant to June 1, 1970.

19.	August 4, 1969	Request to applicant for additional information on site and environment, reactor coolant system, containment system, engineered safety features, instrumentation and control, electrical systems, waste disposal and radiation protection, conduct of operations, and accident analysis.
20.	August 22, 1969	AEC-DRL staff requests copies of monitoring reports and status of actions on Fish and Wildlife recommendations.
21.	August 23, 1969	ACRS Subcommittee meeting on tornado protection, emergency planning, permanent incore instrumentation, adequacy of onsite emergency power, and containment isolation.
22.	September 24, 1969	Meeting with applicant to discuss Westinghouse presentation on power distribution detection and control in Indian Point 2.
23.	October 13, 1969	Submittal of Amendment 10 (Supplement #1) responses to AEC regulatory staff's request of March 5, 1969, on medical plans and partial answers to AEC regulatory staff's request for additional information of August 4, 1969.
24.	October 24, 1969	Submittal of Amendment No. 11, replacement pages and responses to AEC regulatory staff's request for additional information of August 4, 1969, on Sections 1, 4, 5, 6, 7, 12, and 14 of the FSAR.
25.	November 13, 1969	Request for additional information on reactor, reactor coolant system, containment system, engineered safety features, auxiliary and emergency systems, initial tests and operations, and accident analysis.
26.	November 21, 1969	Submittal of Amendment No. 12, additional and replacement pages to be inserted into the FFDSAR and further responses to AEC regulatory staff's request for additional information of 8/4/69 on Sections 1, 4, 7, 8 and 11 of the FFDSAR.

27.	December 10, 1969	Meeting with applicant to review electrical drawings including AC power, DC power, Reactor Protection System, and Engineered Safety Features.
28.	December 30, 1969	Meeting with applicant and Westinghouse Electric Corporation to continue detailed review of electrical drawings including Reactor Protection System and Engineered Safety Features.
29.	January 16, 1970	Meeting with applicant to review and discuss electrical drawings including Reactor Protection System and Engineered Safety Features.
30.	January 21, 1970	Meeting with applicant & Westinghouse Electrical Corporation on technical specifications.
31.	January 27, 1970	Submittal of Amendment No. 14, replacement pages for FSAR & further responses to AEC-DRL questions of 8/4/69 & 11/13/69, chapters 1, 4, 6, 11, 12 & 14.
32.	February 17, 1970	Meeting with applicant for presentation of results of Con Ed's Analysis concerning potential damage to Indian Point 2 and IP-3 from a failure of the IP-1 superheater stack.
33.	March 2, 1970	Submittal of Amendment No. 15, responses to AEC regulatory staff's requests for additional information of 8/4 and 11/13, 1969 and Containment Design Report.
34.	March 10, 1970	Request to applicant for additional financial data.
35.	March 13, 1970	Meeting with applicant to discuss questions concerning core heat transfer and burnout limits, fuel element performance and ECCS performance during a LOCA.

36.	March 19, 1970	Meeting with applicant, Westinghouse presentation on iodine removal system for IP-2.
37.	March 26, 1970	Meeting with applicant to discuss analysis of fresh water flood and changes to electrical systems.
38.	March 30, 1970	Submittal of Amendment No. 16, additional and replacement pages for the FSAR and further responses to the AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
39.	April 25, 1970	ACRS Subcommittee meeting and meeting with applicant on instrumentation and control, and anticipated transients with failure to scram.
40.	April 17, 1970	Submittal of Amendment No. 17, additional and replacement pages to be inserted into the FSAR and further responses to AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
41.	April 29, 1970	Meeting with applicant to discuss seismic and structural design questions for IP-2.
42.	May 5, 1970	Meeting with applicant to discuss failure mode analysis of the engineered safety feature manual actuation panel.
43.	May 11, 1970	ACRS Subcommittee meeting at the Indian Point 2 site to discuss instrumentation and control and Electrical Systems.
44.	May 12, 1970	AEC issued Order extending completion date for construction of the IP-2 plant to June 1, 1971.
45.	May 28, 1970	ACRS Subcommittee meeting to discuss loss-of-coolant accident, anticipated transients with failure to scram.
46.	June 3, 1970	Submittal of Amendment No. 18, additional and revised pages for the FSAR in response to AEC regulatory staff request for additional information.

47.	June 11, 1970	ACRS full Committee meeting to consider design of engineered safety feature manual actuation panel and operation with less than four loops.
48.	June 17, 1970	Meeting with applicant to discuss consequences of turbine missiles, sensitized stainless steel control room accident dose, hydrogen recombiner.
49.	July 15, 1970	Submittal of Amendment No. 19 (Supplement 10), additional and revised pages for the FSAR and Flooding Evaluation report.
50.	July 20, 1970	Submittal of Amendment No. 20, (Supplement 11) proposed Technical Specifications.
51.	July 24, 1970	Request for additional information on emergency core cooling, reactor coolant system, instrumentation and control, electrical systems, conduct of operations and accident analysis.
52.	July 28, 1970	Submittal of Amendment No. 21, Con Ed Annual Report.
53.	July 28 and 29, 1970	ACRS Subcommittee meeting to discuss technical specifications, flood protection, Unit No. 1 superheater stack failure and containment sprays.
54.	July 30, 1970	Submittal of Amendment No. 22, (Supplement 12), revised pages for FSAR in response to request for additional information.
55.	August 7, 1970	Meeting with applicant to discuss technical specifications.
56.	August 13, 1970	ACRS full Committee meeting to discuss the matters addressed in our July 2, 1970 report.
57.	August 14, 1970	Submittal of Amendment No. 23 (Supplement 13), answers to request for additional information issued July 24.

58.	August 18, 1970	Meeting to discuss licensed operator requirements.
59.	August 28, 1970	Submittal of Amendment No. 24 (Supplement 14). Revised pages to the FSAR.
60.	September 1, 1970	Meeting with applicant regarding performance of Emergency Core Cooling System.
61	September 9, 1970	Meeting with the applicant to discuss Technical Specifications.
62.	October 21, 1970	Request to applicant for a report on analysis of laminations in base plate material of the IP-2 pressurizer.
63.	October 29, 1970	Meeting with applicant to review technical specifications for the Indian Point 2 plant.
64.	November 1970	Submittal of Amendment 25 (Supplement 15), changes to technical specifications and to FSAR.

APPENDIX B

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

SEP 23 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consolidated Edison Company of New York, Inc., for authorization to operate the Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the Committee's 95th, 98th, 122nd, and 124th meetings, and at Subcommittee meetings on August 23, 1969, March 13, 1970, April 25, 1970, May 28, 1970, July 28-29, 1970, and September 15, 1970. Subcommittees also met at the site on December 28, 1967 and May 11, 1970. The Committee last reported on this project to you on August 16, 1966. During the review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Indian Point site is located in Westchester County, New York, approximately 24 miles north of the New York City limits. The minimum radius of the exclusion area for Unit No. 2 is 520 meters and Peekskill, the nearest population center, is approximately one-half mile from the unit. Also at this site are Indian Point Unit 1, which is licensed for operation at 615 MWt, and Unit 3, which is under construction.

The applicant has re-evaluated flooding that could occur at the site in the event of the probable maximum hurricane and flood, in the light of more recent information, and has concluded that adequate protection exists for vital components and services.

Additional seismic reinforcement being provided for the Indian Point Unit No. 1 superheater building and removal of the top 80 ft. of the superheater stack will enable the stack to withstand winds in the range of 300-360 mph corresponding to current tornado design criteria. Since

the reinforcement of the superheater building, which supports the stack, enables the stack to resist wind loads of a magnitude most likely to be experienced from a tornado, the Committee believes that removal of the top 80 ft. of the stack, to enable it to resist the maximum effects from a tornado, may be deferred until a convenient time during the next few years, but prior to the commencement of operation of Indian Point Unit No. 3. The applicant has stated that truncation of the stack will have no significant adverse effect on the environment.

The Indian Point Unit No. 2 is the first of the large, four-loop Westinghouse pressurized water reactors to go into operation, and the proposed power level of 2758 MWt will be the largest of any power reactor licensed to date. The nuclear design of Indian Point Unit No. 2 is similar to that of H. B. Robinson with the exception that the initial fuel rods to be used in Indian Point Unit No. 2 will not be prepressurized. Part-length control rods will be used to shape the axial power distribution and to suppress axial xenon oscillations. The reactor is designed to have a zero or negative moderator coefficient of reactivity, and the applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

Unit 2 has a reinforced concrete containment with an internal steel liner which is provided with facilities for continuous pressurization of weld and penetration areas for leak detection, and a seal-water system to back up piping isolation valves. In the unlikely event of an accident, cooling of the containment is provided by both a containment spray system and an air-recirculation system with fan coolers. Sodium hydroxide additive is used in the containment spray system to remove elemental iodine from the post-accident containment atmosphere. An impregnated charcoal filter is provided to remove organic iodine.

Major changes have been made in the design of the emergency core cooling system as originally proposed at the time of the construction permit review. Four accumulators are provided to accomplish rapid reflooding of the core in the unlikely event of a large pipe break, and redundant pumps are included to maintain long-term core cooling. The applicant has analyzed the efficacy of the emergency core cooling system and concludes that the system will keep the core intact and the peak clad temperature well below the point where zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The Committee believes that there is reasonable assurance that the Indian Point Unit No. 2 emergency core cooling system will perform adequately at the proposed power level.

The Committee concurs with the applicant that the reactor pit crucible, proposed at the time of the construction permit review, is not essential as a safety feature for Indian Point Unit No. 2 and need not be included.

To control the concentration of hydrogen which could build up in the containment following a postulated loss-of-coolant accident, the applicant has provided redundant flame recombiner units within the containment, built to engineered safety feature standards. Provisions are also included for adequate mixing of the atmosphere and for sampling purposes. The capability exists also to attach additional equipment so as to permit controlled purging of the containment atmosphere with iodine filtration. The Committee believes that such equipment should be designed and provided in a manner satisfactory to the Regulatory Staff during the first two years of operation at power.

The applicant plans to install a charcoal filter system in the refueling building to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. This installation will be completed by the end of the first year of full power operation.

The reactor instrumentation includes out-of-core detectors, fuel assembly exit thermocouples, and movable in-core flux monitors. Power distribution measurements will also ordinarily be available from fixed in-core detectors.

The applicant has proposed that a limited number of manual resets of trip points, made deliberately in accordance with explicit procedures, by approved personnel, independently monitored, and with settings to be calibrated and tested, should provide an acceptable basis for the occasional operation of Indian Point Unit No. 2 with only three of the four reactor loops in service. The Committee concurs in this position.

The applicant stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefulness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

The reactor includes a delayed neutron monitor in one hot leg of the reactor coolant system to detect fuel element failure. Suitable operability requirements will be maintained on the several sensitive means of primary system leak detection.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

The applicant stated that existing experimental results and analyses provide considerable assurance that high burnup fuel of the design employed will be able to undergo anticipated transients and power perturbations without a loss of clad integrity. He also described additional experiments and analyses to be performed in the reasonably near future which should provide further assurance in this regard.

The Committee has, in recent reports on other reactors, discussed the need for studies on further means of preventing common failure modes from negating scram action, and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has provided the results of analyses which he believes indicate that the consequences of such transients are tolerable with the existing Indian Point Unit No. 2 design at the proposed power level. Although further study is required of this general question, the Committee believes it acceptable for the Indian Point Unit No. 2 reactor to operate at the proposed power level while final resolution of this matter is made on a reasonable time scale in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept advised.

Other matters relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS letters should, as in the case of other reactors recently reviewed, be dealt with appropriately by the Staff and the applicant in the Indian Point Unit No. 2 as suitable approaches are developed.

The ACRS believes that, if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit No. 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 2

1. Amendment No. 9 to Application of Consolidated Edison Company of New York for Indian Point Nuclear Generating Unit No. 2, consisting of Volumes I - IV, Final Safety Analysis Report, received October 16, 1968

2. Amendments 10 - 20 to the License Application
3. Amendments 22 - 24 to the License Application

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis Report
Volumes I, II, III and IV dated October 15, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
November 29, 1968

As pointed out in our comments of October 29, 1965 on Unit No. 2, a primary influence on the meteorological statistics of the Indian Point site seems to be its location in a river valley about a mile wide with terrain rising 600 to 1000 feet on either side. Consequently, wind directions follow a pronounced diurnal cycle with daytime, unstable (lapse) flow in the upriver direction and nighttime, stable flow in the downriver directions. The report documents a 42.4 percent inversion frequency, but it should also be pointed out that inversion conditions are largely confined to the nighttime, downriver flow lasting about 12 hours before changing to lapse or upriver flow. Figure 2.6-1, although in terms of average vectors, shows the marked wind reversals at sunset and sunrise and the rather persistent, channeled flow that can occur during the middle of the night (see the mean direction between 0200 and 0800 hours). The mean wind speeds during this persistent period is about 2.5 m/sec which indicates that 50 percent of the time inversion wind speeds could be less than 2.5 m/sec.

In the absence of specific, joint-frequency wind speed and direction persistence data from the site, a reasonably conservative meteorological model would be to assume for a ground release a 1 m/sec wind speed under inversion conditions in a persistent downriver direction for a period of 8 hours. Taking into account the likelihood of a diurnal wind reversal, a very conservative assumption would be to allow the plume centerline to meander over a 22-1/2° arc under the same conditions for the remainder of the 24-hour period. Again, with no specific on-site wind persistence data, the conservative assumption has been made.

The amount of additional atmospheric diffusion because of the building turbulence can be assessed by the virtual point source expression $[(x + x_o)/x]^{1.5}$ as used by the applicant, which for a value of x_o = 430 m

amounts to a factor of 2.5 at the site boundary (520 m) and 1.6 at the low population boundary (1100 m). These values are in close agreement with the method of using a shape factor of 1/2 and a building cross-section of 2000 m^2.

In summary, from data presently available, it would seem reasonably conservative to assume a persistent wind direction for an 8-hour period under inversion conditions and a 1 m/sec wind speed. With the added assumption of a building wake shape factor of 1/2 and a cross-sectional area of 2000 m^2, the resulting 0-8 hr relative concentration would be 6.6×10^{-4} sec m^3 at the site boundary and 3.7×10^{-4} at the low population boundary. From Table 14.3.5-3 one can calculate that the applicant's model for the 0-8 hr period results in an average relative concentration of 4.8×10^{-4} and 2.4 sec m^{-3} at the site and low population boundary, respectively.

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis
Amendment No. 12 dated November 21, 1969, and
Amendment No. 14 dated January 27, 1970

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 17, 1970

The original documentation of the Indian Point site during the period 1955-1957 indicates that at the 100-ft. height the annual prevailing wind direction is from the north northeast and that in the sector from 22.5 to 42.5 degrees the frequency of inversion, neutral and lapse conditions was 6, 2, and 1 percent, respectively. Within this sector, the shortest site boundary is approximately in a direct line through Units 2 and 3 at a distance of 610 and 380 m, respectively, as measured from figure 2.2-2. It is about 500 m from the Unit 1 stack to this common boundary point. The nearest site boundary, regardless of sector, is where the property line intersects the downriver edge of the site. Although this point is at a distance of 580 m from Unit 2, it is not in the most prevalent wind direction by a considerable amount.

To compute the average annual dilution factor we have assumed the frequencies listed above, averaged over a 20-degree sector with a wind speed of 2, 4 and 6 m/sec, respectively, for inversion (Type F), neutral (Type D), and lapse (Type B) conditions. Assuming no building wake effect our results show the applicant's values for Units 1 and 2 to be reasonably conservative. In the case of Unit 3 we compute an average annual dilution factor of 2.9×10^{-5} sec m^{-3} as compared to the applicant's value of 1.6×10^{-5} sec m^{-3}. The only explanation we have for the ESSA value being twice as high is the use of the building wake effect in the applicant's assumptions.

It is our view that the use of the building wake effect in the long-term average diffusion equation, as was done by the applicant, is inappropriate. It does not seem logical that for the same atmospheric conditions the Sutton equation on page Q 11.10-1 for the long-term model gives <u>more</u> credit for building wake effect than the equivalent short-term model on p. Q 11.10-2. For example at $x = 400$ m assuming $x_o = 400$ m and $n = 0.5$, the building wake effect, $[(x+x_o)/x]^{2-n/2}$, for the long-term equation is 3.4 whereas for the effect in the short-term equation, $[(x+x_o)/x]^{2-n}$, the value is 2.8. It is the larger exponent in the former that makes the difference. Also, the fact that one averages in the horizontal dimension over a sector essentially would nullify any added dilution in that dimension because of wake effect.

APPENDIX D

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CERER

21 November 1969

Mr. Roger S. Boyd
Asst. Director for Reactor Projects
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Boyd:

Reference is made to your letters regarding Docket Nos. 50-247, 50-286, 50-342, and 50-343, Consolidated Edison Company of New York's proposed Indian Point Nuclear Generating Units No. 2 and No. 3, and Units No. 4 and No. 5 which are contiguous to Indian Point plant site.

Pursuant with our arrangements, Mr. R. A. Jachowski and Mr. B. R. Bodine of CERC have reviewed all pertinent information contained in the reports from the standpoint of establishment of a design water level. This included the review of the storm surge associated with the Probable Maximum Hurricane (PMH) and wind wave analysis.

We concur with the applicant's finding that the design water level should be 14.5 feet above the mean sea level datum for Units, Nos. 2, 3, 4 and 5. Although this value is acceptable, there are compensating errors in routing procedure employed.

If you have any further questions regarding this matter please let us know.

Sincerely yours,

Edward M. Willis

EDWARD M. WILLIS
Lieutenant Colonel, CE
Director

APPENDIX E

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

SEP 16 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by R. C. DeYoung is a review of the flood information presented in Amendment No. 19 to the Final Safety Analysis Report for Unit No. 2 Indian Point Nuclear Generating Station. It is presumed that the flood levels for all 3 units at the Indian Point Stations will be based on this amendment. Copies of our earlier reviews, for Unit No. 2 (Aug. 15, 1966) prepared by E. L. Meyer, and for Unit No. 3 (January 6, 1969) prepared by P. J. Carpenter, are attached.

This review was prepared by P. J. Carpenter and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

W. A. Radlinski

Acting Director

Enclosures

Consolidated Edison Company of New York Inc.
Indian Point Nuclear Generating Station Unit No. 2
Docket No. 50-247

The probable maximum flood as defined by the U.S. Army Corps of Engineers, at the site, has been calculated as 1,[illegible],000 cubic feet per second. This discharge is approximately three times greater than the maximum observed flood at Green Island, and is approximately twice the maximum discharge observed for nearby [illegible]-sized drainage basins which appear to exhibit similar runoff characteristics. The stage for the maximum probable flood at the site, computed using standard step-backwater procedures, is given as varying between 13.4 and 14.0 ft msl (mean sea level) depending on concurrent tide levels at the Battery. It is shown that none of the dams on the Hudson River and its tributaries would fail during the probable maximum flood. The above results were obtained using conservative assumptions and appear to be reasonable.

The analyses show that the occurrence of the probable maximum flood on Esopus Creek would cause failure of Ashokan Dam some 75 miles upstream of the site. To establish a flood design level at Indian Point various combinations of the following factors were considered: 1) the flow resulting from the Ashokan Dam failure, 2) various concurrent Hudson River flood flows, and 3) various concurrent tide levels at the Battery. The results of these combinations of factors were compared with the stage of the probable maximum flood (14.0 ft msl) and the stage resulting from the probable maximum hurricane plus spring high tide (14.5 ft msl). The most critical combination investigated consisted of the flows from the Ashokan Dam failure caused by the probable maximum flood on Esopus Creek, the concurrent standard project flow (one half the probable maximum flood), the concurrent stage at the Battery corresponding to the standard project hurricane tide level and wind waves of one foot at the site. This stage is given as 15.0 ft msl. The lowest floor elevation of Unit No. 2 is given as 15.25 ft msl.

Other combinations of the above-mentioned factors, such as Ashokan Dam failure and the standard project hurricane or floods larger than the standard project flood on the Hudson River, could produce higher stages at the site. Depending on the degree of conservatism desired, any of these higher stages could also be selected as the design flood level. However, the stage for the combination selected for the design flood level exceeds those given for the probable maximum flood or probable maximum hurricane when these are considered as independent events.

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

APPENDIX F

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Consolidated Edison Company of New York, Inc.

Docket No. 50-247

by

N. M. Newmark
and
W. J. Hall

Urbana, Illinois

20 August 1970

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REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Indian Point Nuclear Generating Unit No. 2 for which application for a construction permit and an operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles N of the New York City boundary and 2.5 miles SW of Peeksill, New York.

This report is based on a review of the Final Facility Description and Safety Analysis Report (Ref. 1) and the containment design report (Ref. 2). The report also is based in part on the discussion and inspection resulting from the visit to the site on 2 May 1969 by N. M. Newmark and W. J. Hall in conjunction with Mr. K. Kniel and Mr. M. McCoy of AEC-DRL. A number of topics were discussed with the applicant and his consultants at the time of this visit, and subsequently additional information has become available through supplements to the FSAR and through discussions with the personnel of DRS, DRL, and the applicant and his consultants. A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of August 1966 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered there.

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The design criteria for the containment system and Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration coupled with other appropriate loadings to provide for containment and safe shut down. The plant was also to be designed for an Operating Basis Earthquake of 0.1g maximum horizontal ground acceleration simultaneously with the other appropriate loads forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Dynamic Analyses

(a) Containment Building. The answer to Question 1.9 of the FSAR indicates that only the containment building, the primary auxiliary building, and the electric cable tunnel were designed with the use of semi-formal dynamic analyses. A description of the method of analysis employed is given briefly in Section 5.1.3.8 of the FSAR and in Section 3.1.5 of the containment design report. The procedure employed involved a calculation of the fundamental frequency and mode shape by use of a modified Rayleigh method. The base shear for the structure was computed from the period and the spectral response corresponding to the appropriate degree of damping. The base shear was then applied as a loading to the structure as an inverted triangular loading. The shears at the nodes were used to calculate the moments and displacements at various points in the structure. For the structures involved it is believed that the approach leads to a design which is reasonably adequate.

A similar approach was followed for the primary auxiliary building as described in the answer to Question 1.9. It is noted there that a one-third increase over working stress was allowed in the design of the bracing in the

case of the Design Basis Earthquake. This stress is below yield, and it is believed that the design will prove to be satisfactory.

(b) Other Buildings and Equipment. The discussion presented in answer to Question 1.9 of the FSAR for other buildings and equipment such as the control building, fan house, intake structure, etc., indicate that a refined static approach was used, which involves employing the peak value from the appropriate response spectrum curve for a given value of damping and multiplying this by the appropriate mass to obtain the inertial loading. From the description given for the various buildings and items of equipment, and the modeling techniques employed, it is concluded that the inertial loadings used in design are reasonably close to those that might be obtained with a more sophisticated analysis and lead to reasonable design values.

The submission in Question 1.3 of Supplement 13 indicates that the Turbine Building, and Fuel Storage Building Structure above the Fuel Storage Pit were reanalyzed by a multi-degree-of-freedom modal dynamic analysis method to check their adequacy. As a result of this reanalysis, the applicant advises that certain structural modifications will be made to columns and cross bracing in the Turbine Building to insure that it can withstand the DBE. The superstructure of the fuel storage building was ascertained to be adequately designed, without modification to withstand the effects of the DBE. The applicant states that reanalysis of the strengthened turbine building and superheater building for Indian Point No. 1 does not significantly affect the responses calculated for the original structures.

(c) Piping Analysis. The method used by the applicant for analysis of the piping, as described in the answer to Question 1.6 of the FSAR, is the same as was used in Ginna. The peak ground response spectrum value for 0.5 percent damping was used, applied as static accelerations in each direction

separately, and the resulting stresses superposed. It was assumed by the applicant that the piping was supported along rigid systems and therefore not subjected to amplified ground motion at points of support. The system was analyzed with the anchors and supports as actually used, according to the discussion presented to us during the time of our visit in May, 1969. It was the view of the applicant that the thermal motions were greater than any differential ground displacements and the latter therefore are not critical items in the design. In answer to Question 1.13 (Suppl. 13) the applicant advises that relative seismic displacement was considered for the main steam lines, where the largest relative displacements are expected; stress differentials of less than 10% resulted. Also, seismic supports installed to date are those specified in the design and employed in the analyses; where deviations in supports must occur, reanalysis will be carried out. These results and approaches appear satisfactory to us.

Since this plant was designed before recent developments and changes in piping design specifications, the 1968 ASME Addenda were not applied. Blow-down and earthquake were considered as separate items and not combined in this design. We are advised that the response to Question 1.9 of Supplement 12 states that a review of the Indian Point 3 reactor coolant system which is identical to Indian Point 2, for combined earthquake and blow-down indicates that the design is adequate.

It is stated in the answer to Question 1.6 of the FSAR that the approach resulted in a seismic design load approximately equal to 0.60W horizontally and 0.40W vertically taken simultaneously. It is further stated that for the Design Basis Earthquake the sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code

allowable stresses. In a similar manner the stresses in the pipe supports and hangers were limited to 1.2 times code allowable stresses.

The applicant originally made use of the maximum spectrum value only and no modal analyses were made; in other words only a static analysis with uniform accelerations was made. Consideration was not given to modified distribution of the inertial loading to take account of the combination of modal effects.

The response to Question 1.9 of Supplement 8, describing more detailed analyses of the reactor coolant system, feedwater lines, surge lines and typical steam lines by more formal methods as carried out later lends confirmation to the adequacy of the design. On this basis, there is reason to believe that the design is adequate.

Backfill Surrounding Containment Vessel

Nine feet of crushed rock backfill was placed between the external wall of the reinforced concrete containment vessel and the retaining wall holding back the rock on the uphill side. This crushed rock backfill is drained at the bottom to avoid water pressure against the containment structure. The fill is approximately 60 to 70 feet higher on one side of the structure than on the other because of the slope of the rock surface. The design, as discussed in Section 3.1.5 of the containment design report, considered local inertial forces of loose rock as an added loading against the containment pressure vessel, and also considered passive pressures caused by failure of the rock along the surface behind the retaining wall. The localized loadings from these forces were considered in the design of the containment structure and the discussion presented in the containment design report provides reasonable assurance that the containment vessel is capable of resisting these localized forces.

Class I Equipment in Structures other than Class I

The turbine building is Class III and not designed for earthquake loadings. The answer to Question 1.3 of the FSAR indicates that the only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the control building, main steam piping and feedwater piping, all of which could possibly be endangered by the Class III turbine building. It is further indicated there that no special provisions have been provided for protection except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the turbine building, it is not anticipated by the applicant that there will be any structural failure in this area. Our judgment as to the adequacy of this aspect of the design is based on the statement given in the application. And, in this respect, the answer to Question 1.3 (Supplement 13) which describes the analysis and strengthening of the Turbine Building and Superheater Building for Indian Point Unit No. 1, and their ability to withstand the DBE, should give additional protection for the control room.

It is further stated that the only Class III crane whose failure could endanger any Class I function is the fuel storage building crane and that the failure of this crane will not impair a safe and orderly shutdown. The answer to Question 1.3 (Suppl. 13) indicates that the only potential for crane lift off will be in the unloaded condition with the trolley parked near the support; the applicant advises that the unloaded crane will not be parked over the pool, so no hazard exists. It is also noted in the answer to Question 1.1.3 that the manipulator crane in the containment building,

a Class III crane, is restrained from overturning and will not endanger Class I structures.

Deformation Criteria

The general stress criteria applicable to the seismic design are summarized in Appendix A of the FSAR. The statement given on page A3 of Appendix A states that for all components, systems and structures classified as Class I, the primary steady state stresses, when combined with seismic stresses resulting from the response to the Design Basis Earthquake, are limited so that the function of the component system or structure shall not be impaired so as to prevent a safe and orderly shut-down of the plant.

We were advised at the time of our inspection of the plant in May 1969 that, for normal loadings plus the Operating Basis Earthquake, the intention was to use code allowables plus the 20 percent increase for transient conditions on Class I components and systems. For the Design Basis Earthquake and blow-down, basically the same criteria were used, although originally it had been planned to adopt higher allowables going into the plastic range using the code for faulted conditions. In actuality, as described in the answer to Question 1.7 of the FSAR, the allowable stresses in the case of the Design Basis Earthquake were limited to the yield point, or slightly below (see answer to Question 1.3 of Supplement 13).

The only references that we note where there was a calculation of stresses exceeding the yield point were at several places in the containment design report where it was mentioned that the calculations indicate that there could be possible local yielding of the liner under certain loading combinations, but that this would be limited and not be expected to be of a nature as to cause concern with regard to the integrity of the liner.

Reactor Internals

The mechanical design and evaluation of the reactor core and internals is described generally in Section 3.2.3 of the FSAR. From the discussion given it appears that the core support structure and core barrel have been designed with proper attention to support points and limitations of motions. The design criteria for the internals themselves, and specifically with reference to deflections under abnormal operation, are given in Table A.3-2 of the FSAR. These appear reasonable and should provide an adequate margin of safety.

Large Penetrations

A finite element analysis of the large penetrations in the containment vessel was made by the Franklin Institute and a description of the analysis and the results obtained is presented in the containment design report. Several analyses were made for different load combinations, and in addition a number of hand calculations were made to check the order of magnitude of the expected forces and stresses and to verify that the results were reasonable Our review of the material presented, to the extent possible, indicates that the penetration design is adequate.

Splices in Large Reinforcing of Bars

Cadweld splices were used in general in the construction of the containment vessel. We were advised that the early splices, about 10 percent of the total, were made with a bronze base, and the remaining 90 percent were made with ferritic base filler metal. Around the hatch opening, we observed that there was approximately a three foot stagger of adjacent splices, but in questioning we learned that there may not be such a stagger over other areas of the containment vessel. Lack of stagger of adjacent splices could

lead to planes of weakness and cause cracking under conditions of over-loading. The pressure tests, however, will reveal any such cracking.

Approximately one in 200 splices was removed for test purposes. This is generally adequate.

Instrumentation and Controls

At the time of the May 1969 visit it was ascertained that the applicant considers the control room as a Class I structure and intends that the housing of it will also be subject to Class I requirements. However, the instrumentation for the control room as well as other instrumentation critical to containment and safe shutdown, has been purchased from the vendors according to applicant's specifications. The answer to Question 1.9 describes the vibration tests employed for selected items of essential equipment; the purpose of these tests is to help demonstrate that little or no difficulty will be expected in the operating characteristics thereof under seismic conditions. Although not absolute proof of acceptability, satisfactory test results certainly help to confirm the adequacy of such instrumentation and control items. Further information on the design and procurement approach for protection system equipment is given in the answer to Question 7.27 (Suppl. 13), and lends confirmation to the approach adopted.

Tornado Loadings

The information contained in Section 3.4 of the containment design report, and the answer to Question 5.7 of the FSAR indicates that the structure is designed for the usual wind loadings. The analyses described in Appendix B of Supplement 6, indicate that the containment building can resist the design tornado. What effect if any that a tornado could have on the control room or other critical facilities is not stated. However, the applicant states that

the siding of the control room can resist wind velocities up to 162 mph, and the girts (supporting the panels) will fail at 0.62 psi negative pressure; the building is protected by other buildings on the south and west.

Steel Liner and Containment Vessel

The analyses that have been carried out with regard to the liner are summarized in the FSAR and some additional information is presented in the containment design report. It is our understanding that where bulges of the liners occurred during construction, of less than 2 in., nothing was done to correct the bulges. However, when bulges were 2 in. or greater the liner was pushed back into a position of not more than 2 in. away from its intended position, and additional studs were used to anchor the liner in place. Temporary bracing was employed to hold it in position until the concrete was cast. Because of the foregoing, and since the temperature rise in the lower part of the structure in the liner is reduced by the use of insulating material, it is not expected that the departures from the intended original surface will lead to any difficulties.

Proof Test Procedures and Instrumentation

It is our understanding that a detailed description of the proof test procedures is to be submitted at a later date. At the time of our visit in May 1969 it was proposed by the applicant that strain readings be taken only on the liner around the penetrations. We suggested that additional readings be made which would include diameter changes of the penetrations and other measurements that can be made conveniently and without excessive expense to provide evidence that the design meets the design criteria. Fig. 5.13-4 suggests that such readings will be made. In any event, an

interpretative report on the measurements that are taken should be provided and should be correlated with the calculations to provide evidence of validity of the design calculations.

Protection of Pipe Lines for Service Water

We were advised that pipelines for service water are embedded in the ground without any special protection. However, there appear to be alternate lines, although they are generally in the same location and/or trenches. In view of the foundation conditions surrounding the plant, and since there is no indication of previous fault motion or potential faulting, this design approach appears to be adequate. If redundancy in critical water supply is desired, it would be preferable to have separate water lines following independent routes.

Seismograph Installation

The answer to Question 1-1 of Supplement 3 indicates that one seismograph will be installed in the yard area, to provide further evidence of the extent of seismic excitation to which the plant might be subjected if an earthquake occurs. This is acceptable to us.

Containment Design Report

The containment design report, prepared for the applicant by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, has proven to be helpful in arriving at an evaluation of many of the factors inherent in the design. The tables presented are useful in helping to arrive at decisions as to the adequacy of the design; we commend those responsible for the preparation of this summary type material.

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We should like to encourage this type of approach to studies of the containment, structures, piping, equipment and other Class I items. We should like to urge that attention be given also to summaries and tabulation of the most important information, in terms of stresses and deformations, including the sources of the various stress components, how they were combined, and related discussion and explanatory material (including figures) which would lend itself to a much better basis for judgment as to the adequacy of design of nuclear facilities in general.

CONCLUDING REMARKS

On the basis of the information made available to us concerning the Class I structures, piping, reactor internals, and other Class I items, it is our belief that the plant possesses a reasonable margin of safety to meet the original design requirements, including the imposed Design Basis Earthquake loading conditions.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Vols. I through V including Supplements 1, 2, 4, 5, 6, 7, 8 and 13," Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., AEC Docket No. 50-247, 1969 and 1970.

2. "Containment Design Report," for Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., prepared by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, March 1969. (Labeled Final Draft)

3. "Adequacy of the Structural Criteria for Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2," by N. M. Newmark and W. J. Hall, August 1966.

W. J. Hall

APPENDIX G

UNITED STATES
DEPARTMENT OF THE INTERIOR
OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

OCT 1 6 1970

Dear Mr. Chairman:

Pursuant to Section 5 of Public Law 89-605 as amended and other authorizations, we are presenting the views of the Department of the Interior in the matter of the application by the Consolidated Edison Company for an operating license for Indian Point Nuclear Generating Unit No. 2, Buchanan, New York, AEC Docket No. 50-247 (Amendment No. 9). The following comments incorporate those submitted by the Federal Water Quality Administration, the Fish and Wildlife Service and the Bureau of Outdoor Recreation.

The unit under review is the second of three units completed or being constructed at the Indian Point site. We note that applications for construction permits for two more units to be located approximately one mile south of the Indian Point site were made in June 1969.

The Department of the Interior does not object to the issuance of the operating license to the Consolidated Edison Company for Unit No. 2 of the Indian Point Nuclear Power Plant. Our position is based upon the firm commitment by the Company as expressed in its responses to the Atomic Energy Commission that it will meet the water quality standards applicable to the receiving waters and that it will take whatever steps are necessary to mitigate any harmful effects that operation of the plant may have on the fishery resources of the Hudson River and tributary waters.

The Company should be commended for the cooperation it has extended to representatives of this Department during the course of our review. The studies which the Consolidated Edison Company is presently engaged in indicate the Company's concern for the potential damages to the environment that could result from operation of this unit and the others planned at and in the vicinity of Indian Point.

We are pleased to note that the Company has made provisions to open part of its land holdings for compatible public recreation use. We express the hope that the Company's public use plans will be finalized and fully implemented at the earliest possible time.

Consolidated Edison has initiated or participated in a number of studies to determine the effects of both radiological and thermal discharges from the Indian Point reactors upon both the temperature distribution and the aquatic life of the Hudson River through its consultants, Quirk, Lawler and Matusky Engineers, and the Alden Research Laboratories of Worcester Polytechnic Institute. The Company has conducted mathematical studies of the probable temperature in the River and has checked these estimates with hydraulic model studies and actual field studies. In addition, Consolidated Edison has supported several independent but coordinated studies of the micro-organisms and aquatic life in the Hudson River and the probable effects of temperature and salinity changes upon them in the vicinity of the Indian Point Plant.

These studies are continuing and have been and will be helpful in assessing the effects of the Indian Point Unit No. 2 and of the other thermal plants which are proposed for construction on the shores of the Hudson River in the vicinity of Indian Point.

We have been provided information on plans for environmental monitoring of radiological and thermal releases proposed as a part of the operating license application. We understand that the plans for water quality monitoring, including radiological concentrations in the environment in microscopic and macroscopic aquatic life are acceptable to the State of New York. They appear reasonable and are considered generally acceptable to the Department of the Interior.

Through the monitoring programs the Company should have the necessary information to control its activities in a manner that will not violate applicable New York State as well as Federal water quality standards, recommendations of any enforcement conference or hearing board approved by the Secretary or order of any court under Section 10 of the Federal Water Pollution Control Act, and/or other State and Federal water pollution control regulations.

In view of the extensive and valuable fish and wildlife resources in the project area, it is imperative that every possible effort be made to safeguard these resources. Therefore, it is recommended that the Consolidated Edison Company be required to:

1. Continue to work closely with the Department of the Interior, New York State Department of Health, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys as planned. These surveys should include but not be limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the reactor effluent outfall.

   b. Beta and Gamma radioactivity analysis of selected plants and animals (including mollusks and crustaceans) collected as near the reactor effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and provide five copies to the Secretary of the Interior prior to project operation.

4. Conduct post-operational radiological surveys similar to that specified in recommendation (2) above, analyze the data, and prepare and submit reports every six months during reactor operation or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to appropriate State and Federal agencies for evaluation.

In addition to the above, the Atomic Energy Commission should urge the Consolidated Edison Company to:

1. Meet with the Department of the Interior, New York State Department of Environmental Conservation, New York State Department of Health, and other interested Federal and State agencies at frequent intervals to discuss new plans and evaluate results of the Company's ecological and engineering studies;

2. Conduct post-operational ecological surveys planned in cooperation with the above named agencies, analyze the data, prepare reports, and provide five copies of these reports to the Secretary of the Interior every six months or until the results indicate that no significant adverse conditions exist;

3. Construct, operate, and maintain fish protection facilities at the cooling water intake structure as needed to prevent significant losses of fish and other aquatic organisms; and

4. Modify project structures and operations including the addition of facilities for cooling discharge waters and reducing concentrations of harmful chemicals and other substances as may be determined necessary.

We appreciate the opportunity to provide these comments.

Sincerely yours,

Walter J. Hickel

Secretary of the Interior

Honorable Glenn T. Seaborg
Chairman, United States
Atomic Energy Commission
Washington, D. C. 20545

APPENDIX H
CONSOLIDATED EDISON COMPANY OF NEW YORK
DOCKET NO. 50-247
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1969	1968	1965
Long-term debt	$1,981.6	$1,901.6	$1,711.0
Utility plant (net)	3,793.3	3,583.6	3,169.5
Ratio - debt to fixed plant	.52	.53	.54
Utility plant (net)	3,793.3	3,583.6	3,169.5
Capitalization	3,818.4	3,667.6	3,228.1
Ratio - net plant to capitalization	.99	.98	.98
Stockholders' equity	1,836.7	1,766.0	1,517.1
Total assets	4,069.6	3,845.4	3,387.0
Proprietary ratio	.45	.46	.45
Earnings available to common equity	93.1	95.7	89.9
Common equity	1,210.2	1,139.0	1,072.1
Rate of return on common equity	7.7%	8.4%	8.4%
Net income	127.2	128.5	111.8
Stockholders' equity	1,836.7	1,766.0	1,517.1
Rate of return on stockholders' equity	6.9%	7.3%	7.4%
Net income before interest	198.0	193.9	168.4
Liabilities and capital	4,069.6	3,845.4	3,387.0
Rate of return on total investment	4.9%	5.0%	5.0%
Net income before interest	198.0	193.9	168.4
Interest on long-term debt	84.3	77.0	62.7
No. of times fixed charges earned	2.3	2.5	2.7
Net income	127.2	128.5	111.8
Total revenue	1,028.3	982.3	840.2
Net income ratio	.124	.131	.133
Operating expenses (incl. taxes)	830.5	788.3	668.6
Operating revenues	1,028.3	982.3	840.2
Operating ratio	.81	.80	.80
Retained earnings	426.1	400.9	321.7
Earnings per share of common	$2.47	$2.57	$2.42

Capitalization at 12/31	1969		1968	
	Amount	% of Total	Amount	% of Total
Long-term debt	$1,981.6	51.9%	$1,901.6	51.9%
Preferred stock	626.6	16.4	627.0	17.1
Common stock	1,210.2	31.7	1,139.0	31.0
	$3,818.4	100.0%	$3,667.6	100.0%

Moody's Bond Ratings:
First Mortgage Bonds — A

Dun and Bradstreet Credit Rating — AaA1

Supplement to

Appendix K

SUPPLEMENT NO. 1

TO

AEC REGULATORY STAFF SAFETY EVALUATION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY

INDIAN POINT NUCLEAR GENERATING PLANT UNIT 2

DOCKET NO. 50-247

November 20, 1970

Prepared by

Division of Compliance
U. S. Atomic Energy Commission

I. INTRODUCTION

This supplements the Safety Evaluation dated November 16, 1970, prepared by the Division of Reactor Licensing of the Atomic Energy Commission (Commission or AEC) in connection with its review of the application of the Consolidated Edison Company (applicant) for an operating license for Unit 2 of the Indian Point Nuclear Generating Station located in the village of Buchanan, in Westchester County, New York.

The AEC regulatory program is founded in the Atomic Energy Act of 1954, as amended, and on implementing regulations and policies adopted by the Commission. The Congress of the United States has established a system of licensing privately owned and operated nuclear facilities. Inherent in the concept of private activities subject to licensing and regulation by a Government agency is the fact that the licensee is held responsible for meeting the objective of the licensing and regulatory system under the provisions of the Atomic Energy Act of 1954, as amended. The objective of the AEC program is to assure that licensed activities will not be inimical to the health and safety of the public or to the common defense and security.

The Division of Compliance, as an integral part of the Commission's regulatory staff, is responsible for conducting the field inspections of AEC licensees to assure that

licensed activities are in compliance with the provisions of AEC licenses; the Atomic Energy Act of 1954, as amended; and the rules and regulations of the Commission. Division of Compliance inspections of nuclear power reactors under construction pursuant to an AEC construction permit provide the principal basis for findings as to the status of completion of facility construction and the conformity of that construction to the requirements noted above.

The Division of Compliance inspection program is conducted from five regional offices with each office having responsibility for the inspection of all AEC licensed activities within an assigned geographical area. The inspection program at Indian Point Unit 2 is the responsibility of the Division of Compliance, Region I office located in Newark, New Jersey. A senior reactor inspector, who reports to the Regional Director, is responsible for supervising the inspection program carried out by the various reactor inspectors. Technical direction of the inspection program is provided by the Division of Compliance Headquarters staff which gives direction to the region with respect to the conduct of inspection activities, gives technical support to the region when required, keeps the region informed concerning inspection experiences in other regions, evaluates adequacy of inspections and inspection

results, and maintains liaison with other divisions of the AEC regulatory staff on matters which affect the inspection program.

The program for the inspection of the construction of Indian Point Unit 2 has been carried out primarily by Region I inspectors but, in addition, by Division of Compliance Headquarters staff, by other divisions of the AEC regulatory staff, and by consultants to the AEC. The principal activity of the inspectors has involved periodic inspections at the construction site. These site inspections were conducted at non-regular intervals with the inspection frequency dependent on the activities which were in progress at the site. In addition to site inspections, there were inspections at the shops of major equipment suppliers (vendors). There were also inspections at the offices of the applicant and at its contractors for the purpose of inspecting construction records and procedures and engineering reports related to construction matters.

Division of Compliance inspection personnel are experienced and knowledgeable in the practical aspects of construction and operation of nuclear reactors. In addition to the inspectors, specialists in appropriate fields of engineering and technology, who are assigned to the Division of Compliance Headquarters staff and to other divisions of the regulatory staff, are utilized to assist

in special inspections. Further, consultants to the AEC also provide assistance as required. The experience and technical competence of inspection personnel are important factors in the effectiveness of the inspection program.

The Division of Compliance inspection activities were directed toward verifying, on a planned sampling basis, that the licensee carries out his safety responsibilities and that the completed facility would conform to AEC regulatory requirements. Systems and components of the facility were selected for inspection on the basis of the regulatory staff's determination as to their importance to the safe operation of the facility. These inspection activities included the following:

1. Review of the applicant's overall quality assurance and quality control programs and their implementation.
2. Inspection of quality control records such as concrete strength test data, material test reports for plate and piping, supplier certifications for piping, valves and fittings, and nondestructive test records for welding.
3. Observation of construction work in progress; e.g., concrete placement, welding associated with vessel construction or piping installation, equipment alignment and installation, and nondestructive testing.
4. Review of construction procedures; e.g., welding procedures and nondestructive testing procedures.

5. Witnessing the performance of major construction tests such as hydrostatic tests of piping and the pressure test of primary containment.

6. Review of program for functional testing of systems and equipment, including the tests planned, the test procedures, and the test results.

7. Review of preparations for facility operations, including such areas as organization and staffing plans and their implementation, program and procedures for fuel loading and power testing, development of routine operating procedures, maintenance procedures, radiation protection procedures, and emergency procedures.

8. Review of component vendor work in progress, quality control activities and records, and fabrication procedures.

The licensee is required to develop and carry out a comprehensive preoperational testing program. The procedures developed under this program are reviewed by Compliance inspectors and comments are directed to the licensee. The performance of selected preoperational tests are witnessed by Compliance inspectors. The results of the tests and the licensee's evaluations are reviewed by the inspectors. This testing of the plant, to the extent possible prior to the loading of fuel, demonstrates whether plant systems and components are capable of performing their intended functions under both normal and abnormal conditions. These tests

also serve to demonstrate the adequacy of plant design and operating procedures. Satisfactory completion of the pre-operational testing program is an important part of the basis for our findings of plant completion.

II. RESULTS OF CONSTRUCTION INSPECTIONS

Since the issuance of Provisional Construction Permit No. CPPR-21 to the applicant authorizing construction of Indian Point Unit 2, inspections by the Division of Compliance have been conducted at the construction site, at vendor shops, and at the applicant's offices. A chronology of these inspections is attached as Appendix A. The results of the inspection of Unit 2, conducted through October 14, 1970, are discussed by systems in the same order as presented in the Safety Evaluation dated November 16, 1970, prepared by the Division of Reactor Licensing.

A. Reactor Coolant System

1. Reactor Coolant Pressure Piping

The reactor coolant pressure piping includes the four primary recirculation loops, the pressurizer lines and portions of the following systems: Chemical and Volume Control, Emergency Core Cooling (ECCS), Shutdown Cooling, Safety and Relief Valves, and Reactor Coolant Vent and Drain.

Our inspection program was directed primarily toward auditing fabrication, erection, and nondestructive testing of the reactor coolant pressure boundary components and piping. The effort included site and vendor inspections utilizing our staff specialists. The hydrostatic test of the reactor coolant boundary at 125% of design pressure, which is required by the American Society of Mechanical Engineers (ASME) Code, has been conducted. Portions of this test were reviewed and witnessed by Division of Compliance inspectors and records of test results were examined to assure compliance with the code. In addition to the normal quality control inspections, a special quality control inspection was performed, under the direction of the assigned inspector, by a team of staff specialists, a specialist from the Division of Reactor Licensing, and a consultant. Segments of the reactor coolant system and emergency core cooling system (ECCS) were selected for inspection and review. Material certifications for selected portions of the reactor coolant system components were examined.

Onsite quality control records for the reactor coolant and ECCS systems were examined and visual inspections of these systems were performed. Followup inspections have been made to the site to complete the record review, and at the vendor shop which fabricated the ECCS piping.

The applicant and his contractor performed a review of quality control records for all pipe, valves, and fittings within the reactor coolant pressure boundary. This review confirmed the Division of Compliance findings that the reactor coolant system piping had not received the full hydrostatic test required by the applicable American Society for Testing Materials (ASTM) Code prior to leaving the manufacturer's shop and that certain cast valve discs (7) had not been radiographed. The subsequent performance of a field hydrostatic test of the system is considered to fulfill the code requirements. The necessity for radiographing the discs of the seven valves which do not perform a primary isolation function is being evaluated by the Division

of Compliance and the Division of Reactor Licensing.

Completion Status: Construction of the primary coolant piping is essentially complete. Some installation of insulation and pipe hangers remains.

2. Reactor Vessel

The reactor pressure vessel was fabricated at the shops of Combustion Engineering, Inc., in Chattanooga, Tennessee.

The Division of Compliance performed inspections at the shops during which fabrication practices were observed, material quality records were examined, and nondestructive testing methods were reviewed. We have followed the placement of the vessel and fitup of the reactor core internals and installation of the internals vibration detection instrumentation. No deficiencies were identified.

Completion Status: Construction of the reactor pressure vessel and core internals has been satisfactorily completed.

3. Steam Generators

Compliance performed a vendor inspection at the steam generator manufacturer's plant.

This inspection included a review of quality control programs and related essential documentation. The inspection disclosed records which indicated that insulation nut plate welds on the channel heads of the steam generators had not been magnetic particle tested. Subsequent magnetic particle testing of the welds was performed in the field. The Division of Compliance reviewed fitup and girth welding of the steam generators in the field. This activity included a review of welding procedures, welder qualifications, and weld material certification.

Completion Status: Construction of the four steam generators has been satisfactorily completed.

4. Reactor Coolant Pumps

The reactor coolant pumps have been installed and have received an initial operation checkout. We verified the pump materials and nondestructive testing performance for the reactor coolant pumps during the special quality control inspection referred to in paragraph II. A. 1. of this report.

Completion Status: Construction of the reactor coolant pumps has been satisfactorily completed.

5. Pressurizer

The pressurizer has been installed. We reviewed installation of the vessel and verified that the code stamp indicated construction to applicable codes and regulatory requirements. During pre-service ultrasonic testing of the pressurizer welds, nonmetallic inclusions in the base plate material were detected. The applicant conducted additional nondestructive testing and technical reviews pertaining to the existing condition and concluded that a series of nonmetallic inclusions exist within the base plate material and that laminar defects beyond that allowed by the ASME Section III code do not exist. The applicant has submitted a report on this subject to the Division of Reactor Licensing. The acceptability of these nonmetallic inclusions is under evaluation by the Division of Compliance and the Division of Reactor Licensing. This issue will be resolved prior to licensing.

Completion Status: Construction of the pressurizer has been completed; however, satisfactory resolution of the above base plate material question will be required prior to licensing.

6. Pressure Relief and Safety Valves

We have verified that the pressure relief and safety valves were installed and were set at the vendor shop to relieve at the designated pressure.

Completion Status: Installation of these valves has been satisfactorily completed.

Conclusions: Based on the results of previous inspections and corrective actions taken by the applicant and contractor to date, we conclude that there is reasonable assurance that the reactor coolant system will be completed in accordance with AEC regulatory requirements.

B. Containment and Class I Structures

1. Primary Containment

The primary containment is a steel-lined reinforced concrete structure which houses the reactor coolant system. Our inspection program included selective examination of field

fabrication procedures, observation of field fabrication activities, observation of non-destructive testing, and selective examination of onsite quality control records.

Problems identified by the applicant during construction of the primary containment included:

a. A marked reduction in cadweld yield strengths was encountered.

b. The nominal diameter of the liner exceeded tolerance limits in some instances.

c. Documentation on pipe penetration bellows materials and weldment quality is only partially traceable.

The applicant and his contractors investigated and resolved to our satisfaction problem a. and b. described above, and have initiated programs for correcting item c. Division of Compliance inspectors followed the progress of the completed investigations during inspections by the applicant at the site, and will follow those that are continuing for item c.

Completion Status: The system will be considered complete following concrete closure of one construction access opening, resolution of

the penetration bellows question, completion of the integrated leak rate test, and installation of the reactor coolant system leak detection equipment.

2. Other Class I Structures

Other Class I (seismic) structures at Unit 2 include the primary auxiliary building, the control room, the fuel storage pool, diesel generator building, and the service water intake structure. Vacuum testing revealed leakage at the welds of the fuel storage pool liner. The applicant and contractors have taken appropriate corrective actions. We have inspected the construction of the other Class I structures from the standpoint of construction practices and concrete quality. No problems were identified.

Completion Status: Construction of the other Class I structures is nearing completion. Items to be completed prior to licensing are:

a. Additional reinforcement of the Unit 1 superheater building (required because of Unit 2 considerations) and the Unit 2 turbine building.

b. Installation of a second completely independent turbine overspeed control.

c. Provisions for alternate charging pump cooling water.

d. Added missile protection for the auxiliary feedwater lines.

Conclusions: Based on our inspections to date, we conclude that there is reasonable assurance that the containment and other Class I structures will be completed in accordance with AEC regulatory requirements.

C. Engineered Safety Features

1. Emergency Core Cooling System (ECCS)

The emergency core cooling system is comprised of a high pressure system, a residual heat removal system, a recirculation system, boron injection tanks, and pressurized safety injection accumulators. We have inspected the construction and examined quality control records for the ECCS during our normal inspections and the special quality control inspection. Results of our inspection included the following:

a. Welding quality control records incomplete.

b. Visual inspection indicated a weakness in first line quality control; i.e., weld splatter, arc strikes, and excessive grinding.

c. Accumulator check valves which were not manufactured to Westinghouse specifications.

The applicant and contractor initiated corrective actions for these items and resolution of each is nearing completion. These items will be reviewed by the Division of Compliance to assure satisfactory resolution prior to licensing.

The applicant and his contractor performed a review of quality records for all pipe, valves, and fittings included in the reactor coolant pressure boundary, as described in paragraph II. A. 1. above. In addition, the applicant has reviewed quality control records for the remainder of the piping included in the ECCS system. The Division of Compliance has audited the results of this review and considers the findings to be acceptable.

Completion Status: Construction of the ECCS system is essentially complete. Remaining work

to be accomplished includes: (1) finish surface cleanup, (2) completion of hanger and support installation, and (3) resolution of items listed above.

2. Containment Spray and Fan Cooling Systems

The containment spray system is comprised of two spray pumps and chemical additive devices. We have inspected the construction and examined quality control records for this system in conjunction with the ECCS.

The containment fan cooling system is located within the containment. The Division of Compliance plans to complete inspection of this system during functional testing and filter testing prior to licensing.

Completion Status: Construction of the containment spray and fan coolers is nearing completion. Work remaining includes filter testing and functional testing.

3. Post Accident Hydrogen Control System

The post accident hydrogen control system has not been installed. Installation of this system will be verified when completed.

Completion Status: Installation of the hydrogen control system will be completed prior to licensing of Unit 2.

Conclusions: Based on the results of our inspections to date, we conclude that there is reasonable assurance that the construction of the Engineered Safety Features will be completed in accordance with AEC regulatory requirements.

D. Instrumentation, Control, and Power Systems

These systems include the reactor protective, control, safety, and nuclear instrumentation and normal and emergency power. We have inspected the quality of the electrical and instrumentation installation, the separation and protection of key safety related circuits, and the loading of cable trays and wireways during the course of our normal inspection and, also, during the special quality control inspection. Our inspection observations included the following:

1. Independent cable design review had not been performed.
2. Independent quality control of cable installation was lacking.

3. Some redundant cables were not properly separated.

4. Items which required additional design analyses.

The applicant and contractor initiated responsive actions to correct the conditions noted above. Compliance has verified that their actions included a 100% design audit relative to the separation of power and control electrical cabling for redundant engineered safety feature and a design review on associated instrument cabling in excess of 95%. We have verified that work on the remaining items listed above is nearing completion. These areas will require additional Compliance inspection effort to assure satisfactory completion prior to licensing.

Completion Status: Construction of the electrical and instrumentation systems is 95% complete. Items remaining to be completed include:

1. Installation of remainder of separation barriers and fire stops.

2. Completion of cable installation surveillance program.

3. Installation of transite barriers at the single penetration area.

4. Installation of redundant power cables for the tunnel fans.

Conclusions: Based on the results of previous inspections and corrective actions taken by the applicant and contractor to date, we conclude that there is reasonable assurance that the instrumentation, control, and power systems will be completed in accordance with AEC regulatory requirements.

E. Radioactive Waste Control

The radioactive waste control system includes facilities for processing and minimizing releases of liquid and gaseous effluents to the environment. We have inspected the installation of the major components of these systems. The radiation monitoring instrumentation has not been installed and will be inspected for acceptable installation prior to licensing.

Completion Status: The radioactive waste control systems are essentially complete with the exception of the radiation monitoring instrumentation and controls.

Conclusions: Based on inspections to date and the applicant's planned actions, we conclude that there is reasonable assurance that the radioactive waste

disposal system will be completed in accordance with AEC regulatory requirements.

F. Auxiliary Systems

Auxiliary systems include chemical and volume control, residual heat removal, component cooling service water, and spent fuel storage.

Completion Status: Construction is essentially complete. Work to be accomplished includes installation of some insulation, hangers and supports.

Conclusions: Based on the results of inspections to date, we conclude that there is reasonable assurance that the auxiliary systems will be completed in accordance with AEC regulatory requirements.

G. Conduct of Operation

Conduct of operation as used here includes organization and staffing, preparation and review of procedures, and the administrative directives which the applicant has developed to conduct the functional testing program and subsequent operation of the Unit 2 facility. We have verified that the applicant has established operational review and audit committees which are actively engaged in activities relating to plant startup. We have verified that the applicant has developed a program

for functional testing of equipment and systems and we have examined the available test procedures on a selective basis. We have also selectively examined the results of tests which have been completed. We have initiated our review of the program and procedures for fuel loading, power ascension testing, and plant operation. We plan to examine these procedures on a selective basis when their preparation has been completed.

Completion Status: Sixty percent of the preoperational test procedures have been approved for use by the applicant. System functional testing is in the initial stages. Preoperational testing, including hot functional testing is scheduled to be completed prior to licensing.

Conclusions: Based on the results of our inspection to date and responsive action taken by the applicant previously, we conclude that the administrative organization is in conformance with the application and that testing will be completed in accordance with AEC regulatory requirements.

III. CONCLUSIONS

Based on the results of inspections of the Indian Point Unit 2 facility, we conclude that construction of

the facility has been substantially completed in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

APPENDIX A

CHRONOLOGY OF COMPLIANCE DIVISION INSPECTIONS
CONSOLIDATED EDISON COMPANY
INDIAN POINT NUCLEAR GENERATING STATION UNIT 2

Date	Type Inspection	Scope of Inspection
5/10-12/66	Shop Inspection - Combustion Engineering, Chattanooga, Tennessee	Inspected shop facilities and discussed procedures for fabricating the reactor vessel.
11/2/66	"	Reviewed fabrication progress of reactor vessel. Observed work in progress and discussed fabrication techniques.
5/2/67	Site Inspection Management Meeting	Initial meeting with Con Ed management to discuss Division of Compliance inspection program during reactor construction.
5/24-26/67	Shop Inspection - Combustion Engineering, Chattanooga, Tennessee	Reviewed fabrication progress, observed work in progress, and inspected records of welding, plate material properties and radiography.
8/1, 16, 22/67	Site Inspection	Reviewed construction organization responsibilities. Inspected containment liner installation. Reviewed quality control program for concrete, reinforcement bar and containment liner activities. The program relating to blasting control was discussed.
11/29-30/67	Site Inspection	Reviewed corrective actions on containment liner bulge. Inspected records on containment liner plate and reinforcement bar materials. Reviewed cadweld splice quality control program and information relating to decrease in cadweld strengths. Inspected concrete compressive strength results. Reviewed blasting control program.

Date	Type Inspection	Scope of Inspection
2/27-28/68	Site Inspection	Reviewed quality control records on cadweld splicing, concrete, containment liner and blasting. Reviewed quality assurance program relative to procurement of off-site components.
4/22-24/68	Vendor Inspection - Combustion Engineering, Chattanooga, Tennessee	Reviewed records of reactor vessel fabrication. Witnessed initial closure of reactor vessel head and hydrostatic testing of the vessel.
3/14/68	Site Inspection	Reviewed quality assurance programs and availability of records for procured components.
6/17-18/68	Site Inspection	Inspected containment liner, cadweld splice, concrete, and blasting records. Reviewed the spent fuel storage liner installation. Inspected steam generator components and reviewed photographs of the steam generator movement from the barge to the site.
6/19/68	Site Inspection	Reviewed vendor inspection reports for procured components. Reviewed purchase specification for the steam generators and the safety injection accumulators.
7/8-9/68	Vendor Inspection Chicago Bridge & Iron, Greenville, Pennsylvania	Reviewed purchasing, quality control, production, and records control for fabrication of the containment liner.
9/27 and 30/68	Site Inspection	Reviewed records pertaining to the containment liner, cadweld splicing and concrete. Reviewed the material receipt inspection program and welding procedures for the safety injection system. Inspected component storage areas. Visually observed the conditions relating to the steam generators and reactor vessel. An initial review of training and preoperational testing was made.

Date	Type Inspection	Scope of Inspection
10/8/68	Site Inspection	Reviewed electrical design criteria relating to cable sizing and tray loading.
11/20-21/68	Site Inspection	Reviewed testing records for cadweld splicing and concrete activities. Reviewed actions taken to resolve quality deficiencies in the conventional and safety injection system pipe. Inspected the reactor vessel, steam generators, and reactor coolant pumps for visible deficiencies.
1/7-9/69	Vendor Inspection - Dravo Corporation, Marietta, Ohio	Inspected fabrication and quality control records pertaining to pipe procured.
1/20 and 24/69	Site Inspection	Reviewed cadweld splicing and concrete test records. Inspected records and procedures pertaining to field fabrication of the reactor coolant system and the steam generator girth welding. Reviewed resolution status of identified conventional pipe deficiencies. Observed machining of the reactor vessel lower internal supports and electrical installation.
3/4-5/69	"	Reviewed records pertaining to cadweld splicing and reactor coolant system welding. Inspected safety injection system weld records and field conditions. Observed steam generator fitup and girth welding and reviewed associated records. Inspected external storage of components.
3/18-21/69	Vendor Inspection - Westinghouse Electric Corporation, Lester, Pennsylvania	Reviewed quality control programs and essential documentation for the steam generators.

Date	Type Inspection	Scope of Inspection
4/22-23/69	Site Inspection	Reviewed pipe specifications, vendor assembly records, storage and installation as related to investigations of piping fabricators' practices.
4/17 and 5/15, 22, 23/69	"	Reviewed quality control records for cadweld splicing, reactor coolant system welding, safety injection system site erection, and the spent fuel pit liner. Reviewed actions taken relative to safety injection and conventional system pipe component deficiencies. Inspected revised steam generator girth weld procedures and records relating to this activity. Reviewed activities associated with pipe fabrication investigations.
6/17, 7/1-2/69	"	Inspected quality control records for cadweld splicing, concrete placement, and welding for the reactor coolant and safety injection systems. Reviewed electrical cable placement control programs and status of investigation relating to pipe procurement. Inspected pipe supports, component outside storage and code stamping of components.
7/23-24/69	Site Inspection	Reviewed progress relating to resolutions pertaining to pipe investigation. Inspected portions of the safety injection system mechanical components to determine proper physical arrangements. Reviewed welder and weld procedure qualification and welding performance for the control rod vessel head seal welds.

Date	Type Inspection	Scope of Inspection
8/26, 27, 29/69 and 9/10/69	Site Inspection	Reviewed the status of the pipe investigation and the proposed organizational changes relating to the establishment of the Wedco, Inc. subsidiary of Westinghouse. Observed reactor coolant system welding. Inspected the electrical cable placement and separations programs. Reviewed the physical layout and preoperational checkout of the fuel storage building. Reviewed procedures for fuel element receipt and storage.
9/30/69 and 10/1-2/69	"	Continued the review of the pipe investigation. Reviewed welding records for the reactor coolant and safety injection systems. Inspected electrical cable placement progress and conformance to separation criteria. Observed the initial receipt and handling of fuel assemblies. Reviewed records relating to containment liner installation at the construction access openings. Reviewed reactor vessel nozzle weld overlay procedures. Observed attachment of reactor vessel internals vibration detectors and control programs for the vessel internals.
12/9-19/69	Quality Control Audit at the site, Con Ed Engineering offices, and Westinghouse Electric Company at Monroeville and Cheswick, Pennsylvania.	Team inspection to evaluate quality control of preselected portions of the reactor coolant, safety injection, main steam, and electrical systems.
2/10/70	Management Meeting	Discussed results of quality control audit performed in December 1969.

Date	Type Inspection	Scope of Inspection
1/22/70 and 2/6 and 11/70	Site Inspection	Reviewed final status of pipe investigation. Inspected the general preoperational test program and initial portions of system flushing and hydrostatic testing procedures. Reviewed conformance to reactor pressure boundary criteria for installed components.
3/26-27/70	"	Continued inspection of preoperational testing program. Reviewed placement and surveillance activities for electrical cables, placement of cadwelds at the containment construction access openings, and status of resolution of items identified during the Quality Control Audit.
4/10, 21, 22/70	"	Continued inspection of the preoperational test program, electrical cable placement, and containment closure. Reviewed the proposed operating organization and status of operator training. Reviewed installation of vibrational detection instrumentation for the core internals.
5/6-8/70	"	Continued inspection of preoperation test program, electrical installation and containment closure. Reviewed status of mechanical surface cleanup.
5/22, 25, 26/70, 6/3, 11, 12, 15, 16/70	"	Continued inspection of the preoperational testing program, electrical installation control programs, mechanical systems cleanup review, and evaluation of reactor pressure boundary components. Made initial inspection of radiation monitoring and waste handling systems.

Date	Type Inspection	Scope of Inspection
6/26 and 29/70, 7/8-9/70	Site Inspection	Witnessed the reactor coolant system hydrostatic test. Continued inspection of preoperational test programs, electrical installation reviews, and previously identified and unresolved items. Made initial inspection of the operating procedure program and nuclear facility safety committee structure and involvement. Reviewed status of previously identified items requiring resolution.
7/30/70 8/4, 5, 19, 24, 25/70	"	Continued inspections of preoperational test programs. Reviewed status of electrical installation, mechanical systems cleanup, reactor pressure boundary, and containment closure activities. Reviewed conditions noted during preservice UT inspection of the pressurizer.
9/8, 23, 25/70	"	Continued inspection of preoperational test program, mechanical system cleanup & containment closure activities. Reviewed installation control programs for pipe supports. Examined ultrasonic test data for the pressurizer base plate material.
10/7, 8, 13, 14/70	"	Continued inspection of the preoperational testing program, mechanical system cleanup, containment closure, and pipe support installation. Reviewed pipe penetration bellows welding and materials documentation. Continued inspection relating to reactor pressure boundary components, electrical design reviews, and electrical cable placement surveillance. Reviewed organization and involvement of the Nuclear Safety Committee. Continued evaluation of the pressurizer base plate material. Reviewed status of previously identified items requiring resolution.