UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.,	)	
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.,	)	License No. DPR 26 and
And ENTERYG NUCLEAR OPERATIONS, Inc.	)	License No. DPR 64
And ENTERGY NUCLEAR OPERATIONS, L.L.C.,	)	
specifically Indian Point Unit 2	)	Docket No. 50-247 and
and Indian Point Unit 3	)	Docket No. 50-286
License Renewal Application	)	

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

Westchester Citizen's Awareness Network. (referred to hereinafter as WestCAN), Rockland County Conservation Association, Inc. (referred to hereinafter as RCCA) Public Health and Sustainable Energy (referred to hereinafter as PHASE) and Sierra Club - Atlantic Chapter (referred to hereinafter as SIERRA CLUB; and New York State Assemblyman Richard Brodsky are individually and collectively referred to hereinafter as Stakeholders, Intervenors, Citizens, or Petitioners), pursuant to 10 CFR § 2.3 09 (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 and License Number DPR-64 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 and Entergy Nuclear Indian Point 3, LLC , and Entergy Nuclear Operations, Inc. (collectively referred to hereinafter as the Applicant, or Licensee, Operator, Entergy; or IP2 LLC and 1P3 LLC; or IP2 and IP3; or Indian Point 2 and Indian Point 3) to renew its operation license No. DPR-26 for Indian Point Energy Center Unit 2 (IP2), and license No, DPR-64 for Indian Point Energy Center Unit 3 (IP3) for twenty years beyond the current expiration date for twenty years beyond the current expiration date of September 28, 2013 and December 12, 2015 respectively.

WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky also request a hearing under 10 C.F.R. §2.309(a).

I. PARTICIPATION AS A MATTER OF RIGHT

A. WestCAN, RCCA, PHASE, SIERRA CLUB and New York State Assemblyman Brodsky have 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).

1. WestCAN has standing on its own behalf and on behalf of its members. WestCAN is a grassroots coalition that has advocated for a nuclear free northeast and has consistently followed the events at Indian in order to keep the public informed through its listserve, WestCAN has approximately three hundred members who live within the State of New York, in Westchester, Rockland, Putnam and Orange County, and who make their residences, places of occupation and recreation within fifty (50) miles of Indian Point, and whose concrete and particularized interests will be directly affected by this proceeding. WestCAN 's central office is located at

2A Adrian Court, Cortland Manor, NY which is within three miles of Indian Point and situated within the Plume Exposure Pathway (EPZ), also referred to as the Peak Fatality Zone.

2. RCCA has standing on its own behalf and on behalf of its members. RCCA is non-profit organization, founded in 1930 and incorporated in 1936. RCCA is dedicated to the conservation of our natural resources, promote sound land use, advocate clean air and water quality, develop proper drainage, support energy conservation and preservation of natural beauty. RCCA has membership of approximately 450, who live within the State of New York, primarily in Rockland, County, and who make their residences, places of occupation and recreation within twenty (20;) miles of Indian Point, and whose concrete and particularized interests will be directly affected by this proceeding. RCCA 's central office is located in Pomona, NY which is within nine miles of Indian Point and situated within the Plume Exposure Pathway (EPZ), also referred to as the Peak Fatality Zone.

3. PHASE as standing on its own behalf and on behalf of its members. PHASE is a grassroots think tank, that advocates for the development and use of sustainable energy, in an effort to protect public health and safety, and the protection of the environment.

PHASE has members who live within the State of New York, primarily in Rockland, Westchester, and Orange Counties and who make their residences, places of occupation and recreation within thirty (30) miles of Indian Point, and whose concrete and particularized interests will be directly affected by this proceeding. PHASE 's central office is located at 21 Perlman Drive, Spring Valley, NY 10977, which is within eleven miles of Indian Point and situated within the Peak Fatality Zone.

4. SIERRA CLUB as standing on its own behalf and on behalf of its members. SIERRA CLUB is North America's oldest, largest and most influential grassroots environmental organization. is a non-profit, member-supported, public interest organization that promotes conservation of the natural environment through public education, lobbying and grassroots advocacy. Founded in 1892, the Sierra Club Atlantic Chapter has more than

45,000 members who are residents New York States. The Atlantic Chapter applies the principles of the national Sierra Club to the environmental issues facing New York State.

SIERRA CLUB has members who live within the State of New York, and throughout the Hudson Valley and New York City, and who make their residences, places of occupation and recreation within two to fifty miles of Indian Point, and whose concrete and particularized interests will be directly affected by this proceeding, many of whom live within the Peak Injury Zone. SIERRA CLUB 's central office is located at 353 Hamilton Street, Albany, NY 12210, with a regional offices in New York City, Dobbs Ferry and Nyack located is within the Peak Ingestion Zone.

5. New York State Assemblyman Richard Brodsky's concrete and particularized interests will be directly affected by this proceeding. His main office is located at 5 West Main Street, Elmsford, NY, located within 15 miles of Indian Point, within the peak "fatality" zone.

Assemblyman Brodsky represents New York State's 92nd Assembly District, which includes the Towns of Greenburgh and Mount Pleasant, the Villages of Ardsley, Dobbs Ferry, Elmsford, Hastings-on-Hudson, Irvington, Pleasantville, Sleepy Hollow, Tarrytown, a portion of the Village of Briarcliff Manor, and part of the City of Yonkers. He serves as Chairman of the Standing Committee on Corporations, Authorities, and Commissions, which oversees the state's public and private corporations.

B. WESTCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky each have standing on their own behalf

1. As stated in Ms. Marilyn Elie's attached declaration, Exhibit A, WestCAN s headquarters are 2A Adrian Court, Cortland Manor, NY within 3 miles of the Indian Point Entergy Center Unit 2 and Unit 3, known as the Peak Fatality Zone.. WestCAN has material archives dating back 10 years.

WestCAN is very concerned that the proposed Indian Point 2, LLC and Indian Point 3, LLC new 20 year superceding licenses that could

increase both the risk and the harmful consequences of an offsite radiological release. Furthermore, WestCAN is concerned that the radiological contamination resulting from radiological releases that 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 WestCAN operations and the residences of its membership. *Id.* WestCAN therefore qualifies for intervention pursuant to 10 C.F.R. § 2.309(d).

WestCAN also qualifies for discretionary intervention. 10 CFR § 2.309(e). WestCAN'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. WestCAN's constituency represents members who have participated in numerous Nuclear Regulatory Commission proceedings and public meetings. The nature of WestCAN's interests is not only its members' (and its own) property interests but the public interest. In particular WestCAN is a lead member of the Indian Point Safe Energy Coalition (IPSEC), a broad coalition of 70 other free standing organizations.

WestCAN can provide local insight that cannot be provided by the Applicant or other procedural parties. WestCAN's members are Indian Point 2 and Indian Point 3's neighbors. In addition, as established in this

proceeding, this proceeding may have significant affect on WestCAN and its members. WestCAN therefore qualifies for discretionary intervention. 10 C.F.R. § 2.309(e). WestCAN is entitled to a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G, because WestCAN has standing, and in the Petition herein to Intervene and Formal Request for Hearing, WestCAN raises substantial issues of fact and law that meet the requirements of 10 CFR §2.310 (d).

2. As stated in Ms. Dorice Madronero's attached declaration, Exhibit B, RCCA's meeting place is at the Pomona Village Hall, on Camp Hill Road, Pomona, New York, within 9 miles of the Indian Point Entergy Center Unit 2 and Unit 3, known as the Peak Fatality Zone.. RCCA has material archives dating back 77 years..

RCCA is very concerned that the proposed Indian Point 2, LLC and Indian Point 3, LLC proposed 20 year superceding licenses could increase both the risk and the harmful consequences of an offsite radiological release. Furthermore, RCCA is concerned that the radiological contamination resulting from radiological releases that 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 RCCA's operations and the

residences of its membership. *Id.* RCCA therefore qualifies for intervention pursuant to 10 C.F.R. § 2.309(d).

RCCA also qualifies for discretionary intervention. 10 CFR § 2.309(e). RCCA's participation may reasonably be expected to assist in developing a sound record. It is well versed in the field of environmental concerns and issues of public health and safety. RCCA's constituency represents members who have participated in numerous Nuclear Regulatory Commission proceedings and public meetings. The nature of RCCA's interest, is not only its members' and its own property interests, but the public interest. In particular RCCA is a member of the Indian Point Safe Energy Coalition (IPSEC), a broad coalition of 70 other free standing organizations.

RCCA can provide local and historical insight that cannot be provided by the Applicant or other procedural parties. RCCA's members are Indian Point 2 and Indian Point 3's neighbors. In addition, as established in this proceeding, this proceeding may have significant affect on RCCA and its members. RCCA therefore qualifies for discretionary intervention. 10 C.F.R. § 2.309(e). RCCA is entitled to a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G, because RCCA has standing, and in the Petition herein to Intervene and

Formal Request for Hearing, RCCA raises substantial issues of fact and law that meet the requirements of 10 CFR §2.310 (d).

3. As stated in Ms. Michel Lee's attached declaration, Exhibit C, PHASE's headquarters are 21 Perlman Drive, Spring Valley, NY 10977 approximately 11 miles of the Indian Point Entergy Center Unit 2 and Unit 3 known as the Peak Injury Zone.. PHASE has material archives dating back seven years.

PHASE is very concerned that the proposed Indian Point 2, LLC and Indian Point 3, LLC proposed 20 year superceding licenses could increase both the risk and the harmful consequences of an offsite radiological release. Furthermore, PHASE is concerned that the radiological contamination resulting from radiological releases that 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 PHASE's operations and the residences of its membership. *Id.* PHASE therefore qualifies for intervention pursuant to 10 C.F.R. § 2.309(d).

PHASE also qualifies for discretionary intervention. 10 CFR § 2.309(e). PHASE'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. PHASE's constituency represents members who have participated in numerous Nuclear Regulatory Commission proceedings and public meetings. The nature of PHASE's interests is not only its members' (and its own) property interests but the public interest. In particular PHASE is a member of the Indian Point Safe Energy Coalition (IPSEC), a broad coalition of 70 other free standing organizations.

PHASE can provide local insight that cannot be provided by the Applicant or other procedural parties. PHASE's members are Indian Point 2 and Indian Point 3's neighbors. In addition, as established in this proceeding, this proceeding may have significant affect on PHASE and its members. PHASE therefore qualifies for discretionary intervention. 10 C.F.R. § 2.309(e). PHASE is entitled to a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G, because PHASE has standing, and in the Petition herein to Intervene and Formal Request for Hearing, PHASE raises substantial issues of fact and law that meet the requirements of 10 CFR §2.310 (d).

4. As stated in Ms. Susan Lawrence's attached declaration, Exhibit D, SIERRA CLUB's headquarters are 353 Hamilton Street, Albany, NY 12210. SIERRA CLUB has material archives dating back to 1892. SIERRA

CLUB's regional chapters have offices in New York City, Dobbs Ferry and Nyack, all located in the Hudson Valley, within the Peak Ingestion Pathway.

SIERRA CLUB is very concerned that the proposed Indian Point 2, LLC and Indian Point 3, LLC proposed 20 year superceding licenses could increase both the risk and the harmful consequences of an offsite radiological release. Furthermore, SIERRA CLUB is concerned that the radiological contamination resulting from radiological releases that would impact the 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 SIERRA CLUB's operations and the residences of its membership. *Id.* SIERRA CLUB therefore qualifies for intervention pursuant to 10 C.F.R. § 2.309(d).

SIERRA CLUB also qualifies for discretionary intervention. 10 CFR § 2.309(e). SIERRA CLUB'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. SIERRA CLUB's constituency represents members who have participated in numerous Nuclear Regulatory Commission proceedings and public meetings. The nature of SIERRA CLUB's interests is not only its members' property interests, but the public

interest. In particular SIERRA CLUB is a member of the Indian Point Safe Energy Coalition (IPSEC), a broad coalition of 70 other free standing organizations.

SIERRA CLUB can provide local insight that cannot be provided by the Applicant or other procedural parties. SIERRA CLUB's members are Indian Point 2 and Indian Point 3's neighbors. In addition, as established in this proceeding, this proceeding may have significant affect on PHASE and its members. SIERRA CLUB therefore qualifies for discretionary intervention. 10 C.F.R. § 2.309(e). SIERRA CLUB is entitled to a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G, because SIERRA CLUB has standing, and in the Petition herein to Intervene and Formal Request for Hearing, SIERRA CLUB raises substantial issues of fact and law that meet the requirements of 10 CFR §2.310 (d).

5. As stated in Assemblyman Brodsky's attached declaration, Exhibit LLL, Mr, Brodsky's office is located at 5 West Main Street, Elmsford, NY, which contain files and records dating back 24 years..

Assemblyman Brodsky is very concerned that the Indian Point 2, LLC and Indian Point 3, LLC proposed 20 year superceding licenses could increase both the risk and the harmful consequences of an offsite radiological

release. Furthermore, Assemblyman Brodsky is concerned that the radiological contamination resulting from radiological releases that would impact the and interfere with his rightful ability to conduct operations in an uninterrupted and undisturbed manner. *Id.* Certainly, any evacuation would severely disrupt and damage Assemblyman Brodsky's operations and therefore he qualifies for intervention pursuant to 10 C.F.R. § 2.309(d).

Assemblyman Broadsky also qualifies for discretionary intervention. 10 CFR § 2.309(e). His participation may reasonably be expected to assist in developing a sound record. It is well versed in the field of nuclear energy and safety. Assemblyman Brodksy has participated in numerous Nuclear Regulatory Commission proceedings and public meetings. As a New York State Assemblyman his interests is not only with regard to his own property interests, but the public interest. In particular Assemblyman Brodsky represents New York's 92nd Assembly District, which includes the Towns of Greenburgh and Mount Pleasant, the Villages of Ardsley, Dobbs Ferry, Elmsford, Hastings-on-Hudson, Irvington, Pleasantville, Sleepy Hollow, Tarrytown, a portion of the Village of Briarcliff Manor, and part of the City of Yonkers. He serves as Chairman of the Standing Committee on Corporations, Authorities, and Commissions, which oversees the state's public and private corporations.

Assemblyman Brodsky can provide local insight that cannot be provided by the Applicant or other procedural parties. In addition, as established in this proceeding, this proceeding may have significant affect on BRODKSY. Therefore Assemblyman Brodsky qualifies for discretionary intervention. 10 C.F.R. § 2.309(e). SIERRA CLUB is entitled to a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G, because Assemblyman Brodksy has standing, and in the Petition herein to Intervene and Formal Request for Hearing, Assemblyman Brodsky raises substantial issues of fact and law that meet the requirements of 10 CFR §2.310 (d).

Although WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky meet the requirements of 10 CFR §2.310(d) for a full adjudicatory hearing on all contentions it raises, WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky do 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 WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky in this case, should that be

necessary to permit WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky to fully adjudicate the important nuclear safety and environmental issues it raises.

C. WestCAN, RCCA, PHASE, and SIERRA CLUB have Representational Standing

Declarations of Ms. Marilyn Elie contained in Exhibit A; and Mr. Mark Jacobs contained in Exhibit D; demonstrate that WestCAN members reside within the immediate vicinity of Indian Point. . WestCAN members also use and enjoy the segment of the Hudson River adjacent to the Indian Point 2 and Indian Point 3 on professional and personal bases. Declarations of Mr. Gary Shaw, contained in Exhibit E; Ms. Jeanne D. Shaw contained as Exhibit F; Judy Allen contained in Exhibit G; and Elizabeth C. Segal contained in Exhibit H.

Declarations of Ms. Dorice Madronero contained in Exhibit B; Ms. Connie Coker contained in Exhibit AAA; Janet Burnet contained in Exhibit BBB; and Andrew Stewart contained in Exhibit CCC, demonstrate that RCCA members reside within the immediate vicinity of Indian Point, live less than 20 miles, and many less than ten miles from Indian Point 2 and 3, 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.;* and that RCCA members also use and enjoy the segment of the Hudson River adjacent to the Indian Point 2 and Indian Point 3 on professional and personal bases.

Declarations of Ms. Michel Lee contained in Exhibit DDD; Ms. Susan Shapiro contained in Exhibit EEE; Robert A. Jones contained in Exhibit FFF, and Maureen Ritter contained in Exhibit GGG, demonstrate that PHASE members reside within the immediate vicinity of Indian Point PHASE's members live less than 20 miles, and many less than ten miles from Indian Point 2 and 3, 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.* PHASE members also use and enjoy the segment of the Hudson River adjacent to the Indian Point 2 and Indian Point 3 on professional and personal bases.

Declarations of Ms. Susan Lawrence contained in Exhibit C; Ms. contained in Exhibit HHH; demonstrate that SIERRA CLUB members reside within the 50 miles of Indian Point,, and are within the Peak Injury Zone, and subject to radiological contamination, evacuation, loss

of property, or other harms in the event of any mishap at the plant. *Id.* Sierra Club members also use and enjoy the Hudson River which is adjacent to the Indian Point 2 and Indian Point 3 on professional and personal bases.

The above declarations of WestCAN, RCCA, PHASE and SIERRA CLUB (collectively referred to as "Stakeholders"), as organizational Intervenors, 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 proceeding. If the proposed new superseding license for Indian Point 2 and Indian Point 3 is granted without first resolving the Stakeholder'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 Stakeholder's members 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 WestCAN, RCCA, PHASE and SIERRA CLUB all have standing in their own right. The issues of public health and safety and environmental protection are germane to WestCAN, RCCA, PHASE and SIERRA CLUB'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. WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky Meet 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 WestCAN, RCCA PHASE and SIERRA CLUB's members, and Assemblyman Brodsky's constituents and furthered by WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky 's purpose.

II. WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky DO 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, WestCAN, RCCA, PHASE, SIERRA CLUB and

Assemblyman Brodsky are 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 WestCAN, RCCA, PHASE and SIERRA CLUB 's members and broad constituency. The contentions submitted herein should not be deeded to waive WestCAN, RCCA, PHASE SIERRA CLUB and Assemblyman Brodsky 's rights to submit further contentions in the future or amend the contentions set forth herein. Further, WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky reserve their rights 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 10CFR §2.102, cross-examination of witnesses will be more efficient when possible for WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky 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.

Stakeholders have 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. WestCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky (the Stakeholders) contend 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 1 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) Memorandum 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 bound under these regulations throughout the License Renewal Application proceedings may not have ex parte communications with the Applicant, with regard to the License Renewal Application.

III. WESTCAN, RCCA, PHASE, SIERRA CLUB and Assemblyman Brodsky SUBMIT FIFTY-ONE 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 (200 1)(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).

CONTENTIONS

CONTENTION #1 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 different 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 meetings has acknowledged that the plants have entirely different histories, different design control and configuration management programs. The NRC

held and continues 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 NRC has assigned separate onsite plant inspection teams to each individual 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.

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 (IP3LLC and Entergy Nuclear Operations ("ENO"), collectively known hereinafter as "ENTERGY".

The Stakeholders contend the Applicant has wrongfully co-mingled and co-joined the license renewal applications for IP2 LLC and IP3 LLC into one application for two separate entities, LLC's and facilities, therefore

the LRA must be denied and sent back to the Applicant for submittal of two unique, independent, site-specific LRA.

CONTENTION # 2 : The NRC routinely violates §51.101(b) in allowing changes to the operating license be done concurrently with the renewal proceedings.

Stakeholders contend the NRC and the Applicant violated §51.101(b) by making, or even entertaining, substantial changes to the operating licenses concurrently with the license renewal proceedings.

On August 2, 2007 the NRC's acceptance of Entergy's LRA was published in the Federal Register (Exhibit Z).

On July 28th, 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. The NRC wrongfully this license transfer application the middle of the relicensing proceedings.

(Contention #3 of this Petition is incorporated by reference, as if fully set forth herein).

On August 16th, While Stakeholders were preparing Intervener Contentions to the License Renewal Application (LRA) which was accepted by the NRC on August 2, 2007, Entergy submitted a modified exemption of fire safety regulations. Without public involvement and in defiance of §51.101(b) the NRC approved the dramatic reduction in fire protection from 1 hour to 24 minutes. The exemption that changes the Current Licensing Basis was published on October 4, 2007.

(Contention #12 of this Petition is incorporated by reference, as if fully set forth herein.)

On September 28th, the NRC granted the exemption to fire protection. The NRC did so, without a public comment period or hearing. The NRC claimed the change from 1 hour to 24 minutes of fire protection, was insignificant, and therefore public comment was not necessary.

On October 4, the exemptions was published in the Federal Registry.

This kind of exemption, which constitutes an operating license amendment, requires 6 and 9 months to be fully evaluated, and often more than a year.

On August 16, 2007 Entergy informed the NRC that the exemption they were requesting was not 30 minutes, but rather only 24 minutes. This was a significant reduction and physically unrealistic to accomplish the necessary analysis and required Safety Evaluation in five short weeks on this brand new issue.

Stakeholders contend that the NRC acted improperly in approved the license amendment/modified exemption request without the required Safety Evaluation. Therefore the exemption must be cancelled.

Stakeholders object to the NRC's grant a finding of no significant hazard with regard to an exemption to the requirements under Federal Rules to be reflected in a forthcoming Safety Evaluation and resulting in an amendment to License No DPR 64 for Indian Point Unit 3, Notice published on October 4, 2007, in the Federal Register. and Stakeholders Petition for Leave to Intervener and Request a Hearing on the above issues, and reopen for consideration the exemption requested due to new, substantial and significant information published on October 4, 2007.

On or about October 2rd, are making rule making changes that allow latitude in terms of fatigue analysis or other forms wear on reactor vessels components that would extensive analysis for an additional 20

years. That under these rulemaking change of thermal shock rule, they would not be required to meet these current standards. Instead they use alternative standards that would reduce safety margins.

These three examples are evidence that the NRC repeatedly violated §51.101(b) , therefore Stakeholder contend that the license renewal process is prejudiced and the LRA cannot be approved.

CONTENTION 3 : The NRC violated its own regulations §51.101(b) 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), some of which do not have a direct relationship with the license.

Issue Statement: Stakeholders assert the ownership and legal liability associated with the proposed new superseding licensing for 20 years is incomplete and inaccurate, as described in Entergy's LRA, by not including holding companies that differ for each plant (see exhibit S, organizational figures). Stakeholders contend the Applicant's request for to the NRC for a transfer of from one LLC after the LRA was already accepted for review by the NRC in Entergy's letter dated July 30, 2007, be denied on the grounds set forth below.

Based upon the documents submitted in the July 30, 2007 letter, the current license does not correctly describe the owners of the Unit 2 & Unit 3 facility, the operators of the Unit 2 & Unit 3 facility are not unambiguous and cause undue confusion of ownership regarding matters relevant to future decisions, especially concerning extended operations during the proposed new superseding license period.

Stakeholder's contend that 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 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.

The structure through which Entergy owns the Indian Point 3 is even more complex because each of the LLCs that owns IP3 is, in turn, 50 percent owned by two other indirect Entergy subsidiaries, Entergy Nuclear New York Investment Company I and Entergy Nuclear New York Investment Company II. These two Entergy Nuclear New York Investment Companies are themselves subsidiaries of Entergy Nuclear Holding Company #1 which, in turn, is a wholly-owned subsidiary of Entergy Corporation. Another Entergy subsidiary, Entergy Nuclear Operations, Inc. ("ENO") operates Entergy's nuclear units in the Northeast. Additional services are provided by other Entergy subsidiaries such as Entergy Services, Inc. (management, administrative and support services) and Entergy Nuclear Fuels Company (nuclear fuel planning, procurement and related services).

Entergy has provided the following explanation for this tiered holding company structure:

> Entergy Nuclear Holding Company, a first tier of Entergy Corporation, has been established with the intent that it will ultimately

hold all the subsidiaries associated with Entergy's nuclear operations. This will consolidate all of Entergy's unregulated nuclear operations under a single holding company, while still supporting the operational and financing demands of the individual plants. *The use of holding companies below Entergy Nuclear Holding Company allows Entergy to segregate various types of financing, investment and business activities, and by doing so, enables Entergy to better manage and control risks associated with these activities.* (Exhibit V page 8)

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 Insecurity* 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 (Exhibit S).

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.

Stakeholder assert that this overly complicated corporate structural overlay has severe consequences to reasonable assurances of health and safety of the public.

Entergy's history regarding its corporate responsibility can be best understood in the aftermath of Katrina, Entergy New Orleans, a subsidiary of 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 – Corp. Watch, Entergy Holds New Orleans Hostage).

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.

Moreover, Stakeholders allege that 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 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.

Therefore, Stakeholders assert that the NRC must deny Entergy's single LRA filed under three distinct companies. In addition, Stakeholder contend that the NRC deny Entergy's license transfer request during the LRA review process, as it clouds the regulatory process.

CONTENTION 4: The exemption granted by the NRC on October 4, 2007 reducing Fire Protection standards are Indian Point 3 are a violation of §51.101(b), and do not adequately protect public health and safety.

Stakeholders contend that the NRC grant of Entergy's modified exemption request to reduce fire safety standards for Indian Point 3, from 1 hour to 24 minutes, approved by on September 28, 2007, and published in the Federal Registry on October 4, 2007.

The granted exemption/license amendment and pending new rules leave the industry less protected from fire than Browns Ferry plant fire was in 1975. Specifically, a fire at Indian Point 3 could cause irreversible loss of control to the reactor loss of power to the emergency cooling systems power cables by one fire in less than 24 minutes. The new exemption from federal

law flagrantly disregards the presidential order for protecting nuclear power against design basis threats, partially codified in 10cfr73.1.

The following list are the laws, rules and orders that were violated in the granting of this exemption.

Laws violated:

APA §557, "initial decisions; conclusiveness; review by agency; submission by parties; contents of decisions; record.

The public participation was forestalled from participation regarding substantial changes submitted on August 16, 2007 by Entergy and not published in the FR.

NEPA consideration of environmental impact from a severe accident mitigation analysis was not done. NRC failed to consider alternative analysis, as is codified in 10CFR50.12. The rule requires detailed alternative analysis which was at best only superficially done—resulting in a conclusion that was incorrect.

Rules violated:

(a) 10CFR §50.57(a)(3) because the exemption does not provide adequate assurance of the health and safety of the public.

(b) 10CFR50.48 and Appendix R because the exemption fails to maintain a fire barrier with a one hour rating. Therefore it is in violation of 10CFR §§54.33(a), and 54.35 and 50.45(h).

(c) 1995 inspection *approved* a firewrap known as HemyC for use in achieving compliance with the one hour Appendix R rule, yet acknowledged in the same report that HemyC was not tested by an approved lab, and that the agency would be following up on it. This approval was a violation of 10CFR50.48 and Appendix R, Section G.III.

(d) Exemption granted with last minute change – it was substantial and therefore should have had public opportunity for involvement. Change was submitted on August 16, 2007, and granted by the Commission on September 28, 2007, and published in the FR as granted on October 4, 2007.

(e) Exemption granted in the middle of the LRA process—violated 51.101(b)

(f) Exemption granted without consideration of 10CFR73.1

(g) Exemption granted that does not provide a prerequisite Environmental Impact Study (EIS approved "no environmental impact" conclusion on Sept 24, 2007) and, again without public involvement.

(h) Exemption granted without the Current License Basis of the plant known—therefore has no basis for assumptions makes any analysis from an unsubstantiated basis meaningless. This makes 10CFR §54.30 no applicable, and the language that governs is in 10CFR54—encompassing all safety standards including 10CFR50 Appendix R. without the CLB there is no way to identify which standards apply and which are in compliance. In addition, 50.54 specifically calls for protection from malevolent acts. (10CFR73.1(a)

(i) Incentive to use discretionary enforcement is abused. The 42 licensees that are forgiven for failure to comply are given both a three year grace period and complete amnesty for failure to comply since Appendix R and 10CFR50.48 were enacted in 1979.

(j) Granting over 900 exemptions to Appendix R, indicates that the federal regulator did not intend that licensees comply with the rule

(k) Awareness of wholesale non-compliance by licensees by the regulator without exemptions even sought indicate that the regulator was not enforcing compliance to the rule on a large scale.

Orders not incorporated in the exemption.

February 25, 2002 Interim Compensatory Measures (ICM) Order (Safety Evaluation related to Order EA-02-026) dated July 17, 2007—where it is noted that real risks of intentional acts will result in more combustible materials then assumed by the plant for the exemption.

Presidential order

On December 17, 2003, President Bush issued Homeland Security Presidential Directive 7 (HSPD-7), which supersedes portions of PDD-63 and clarifies that the Department of Energy is the lead agency with which the energy industry will coordinate responses to energy emergencies.

This exemption raises in scope, material issues of law and facts, supported by facts and expert opinions.

SUMMARY OF ISSUES AND CONTENTIONS

The current license amendment, Indian Point 3 less protected from fire than Browns Ferry plant was in 1975. Specifically, in less than 24 minutes a fire at Indian Point 3 could cause irreversible loss of control to the reactor, and loss of use of the emergency cooling systems power cables. The new exemption from federal law flagrantly disregards the Presidential order for protecting nuclear power against Design Basis Threat, partially codified in 10CFR73.1.

The 1975 fire at the Browns Ferry Nuclear Plant damaged more than 1600 electrical cables and required almost eight hours to contain. It caused loss of ability to control reactor power and to safely shut down the plant during that period. Prior to Brown's Ferry the fire potential of insulation on cables was not considered to be relevant by the industry or the NRC in establishing standards by which nuclear plants should be constructed.

Since then the NRC have reacted with dysfunctional and failed attempts to perform Congress' mandate: " To protect the health and safety of the public". After more than 30 years since the Browns Ferry fire the NRC continues to allow prima facie violations of federal rules by the nuclear industry that directly reduce adequate protection of public health and safety.

By the NRC granting Entergy the exemption on October 4, 2007, they have granted a substantial reduction in Fire Protection Program for Indian Point 3, and condoned the dangerous conditions currently at the facility. This exemption to federal rules, has made Indian Point 3 more vulnerable to fire then Browns Ferry was in 1975. The reduction from a 1 hour fire rating to a 24 minute fire rating, is a significant change in the Current License Basis and Design Basis.

Now, a single fire ignited in an electrical cable [redacted] must be fully detected, responded to by a fire brigade, and fully extinguished in less than 24 minutes, or loss of the control of reactor power will occur, and combined with expected valve openings, will likely cause catastrophic core melt.

Since 1995 the NRC has permitted ongoing violations, and non-compliance by plant operators. This exemption codifies these violations, and permits substantial reduction in defense-in-depth.

The exemption granted did not add in the potential of a deliberate act of sabotage or terrorism, as is required under federal rules mandated after September 11, 2001. The NRC and Entergy failed to consider the act of an insider with specific knowledge of the target, as is required under the Design Basis Threat (DBT), codified in 10CFR73.1.

Under this exemption one individual could set fires in both Unit 2 and Unit 3, causing a melt down both plant, in a matter of hours. This does not require smuggling in the combustibles needed for ignition for sufficient burn time, nor, the act of more than one individual.

The exemption granted on October 4, 2007, only 6 years after 9/11 does not consider ignition of a fire by a light aircraft accidentally or deliberately crashing into the specific and easily identified, [redacted], [redacted], penetrating a two foot wall of concrete, and thus igniting fires. Due to the reduction in fire protection from 1 hour to 24 minutes cables required for safe shut down will be destroyed within 24 minutes

Fire is the single highest threat to plant operational safety.

BACKGROUND

In 1979, four years after Browns Ferry, the NRC enacted new federal regulation intended to strengthen fire protection, however, in spite of new regulation strengthening fire protection standards, the NRC began granting exemption request and after exemption request, for licensee holders.

Over 900 exemptions to date have been granted by the NRC on fire safety. In particular, the one hour rule for suppression without manual

action has been set aside by numerous licensees. Licensees routinely credited manual operator action inside the one hour limit to safely shutdown the plant. Many licensees did not even bother to request exemptions, but simply credited manual actions in the safe shut down procedures thus deliberately setting aside the federal rules.

When the industry lobbied the NRC they adopted a cost benefit analysis disguised as a probabilistic analysis being codified in 10CFR50.48(c), "alternative analysis." Profits of the nuclear industry are now being weighed against protection of public health and safety. Unfortunately it appears that the bias is leaning heavily in favor of corporate profits.

HEMYC fire wrap improperly tested and found to perform for only 24 minutes, instead of 1 hour, as advertised.

In 1995 inspection reports the NRC specifically identified a wire wrap, fire protection, material known as **HemyC as not being properly tested, but accepted by the NRC for protecting electrical tunnels at Indian Point 3.** Full-scale fire tests recently performed by the NRC revealed that HemyC, a fire barrier system used to protect cables in electrical raceways in nuclear power plants, does not perform as

designed. The outer covering of the barrier can shrink during a fire, opening joints in the material and potentially allowing the fire to damage cables inside. These results show that HemyC does not serve as a fire barrier for the full hour required.

Despite these new test that identified that HemyC could not withstand a fire for more than 24 minutes in certain cable set-us, required to be 1 hour it is still be used at Indian Point 3. The NRC issued Generic Letter 2006-03 in April 2006 to ensure that the affected licensees take appropriate corrective actions.

On August 16, 2007, Entergy notified the NRC that deficient design of the HemyC fire wrap would not withstand the originally proposed exemption of 30 minutes, but for an unknown duration with a best guess of 24 minutes --- and that guessed duration would only be *after plant modifications* were completed. The necessary modifications may remain unimplemented up to December 2008.

There was no public comment period . The changes made to the proposed exemption on August 16, 2007 where never made formally public, *almost no one noticed* until after the grant. Even the New York State Attorney General's Office who objected on the same day, believed that the exemption was still pending.

Complete and proper analysis of the implications on fire safety caused by the greatly reduced fire standard usually takes months. However, in a matter of a few short weeks the amended exemption request was accepted by the NRC.

The affect of NRC's grant of the October 4, 2007 exemption, are 1) reduction of fire safety parameters by more than 50%; 2) non-compliance by the operator for more than 10 year, is condoned, despite long term safety violations; 3) failure to consider public comment; and most importantly, 4) erosion of the time available to detect, respond and extinguish a fire that affects both *power* of emergency core cooling systems and the *controls* for those emergency systems and for normal control of reactor criticality itself.

The NRC's public statements regarding fire protection, plant security, and design basis threats are in direct contradiction of the approval of the amended exemption request, in violation of the requirements of 10CFR50.48 and Appendix R.

Congressional Hearings.

The Congressional Energy and Commerce Oversight Committee held a number of hearings questioning the Nuclear Regulatory Commission on the subject of Fire Protection beginning about fifteen years ago. Each NRC

Chairmen listened, accepted responsibility, made commitments, and then failed to act.

Promises by the NRC Chairman Selin in 1993, and by NRC Chairman Meserve in 2001 to the Congressional Energy and Commerce Committee Oversight Committee on Energy and Safety were made independently, 8 years apart and each remain unfulfilled today.

Instead, of fulfilling commitments to improve fire protection compliance to the 1979 rule, the NRC has stripped down the technical basis and fundamental goals of the federal rules regarding fire protection with several initiatives enacting "alternative analysis" to those rules.

There is a substantial record of the NRC's mistakes 1980s and early 1990s, and in more recent hearings in 2004, 2005, and 2006 are obvious. The Nuclear Regulatory Commission was warned in 1993, and then admonished in 2001 for its failure to implement the 1979 rule, and recently questioned again regarding lack of fully implemented rules regarding Design Basis Threat, and the pending rulemaking regarding that by passes the key elements of the 1979 rule completely.

The NRC's failure to enforce the 1979 rules dates back more than 25 years. Portions of the DBT rule, have been side stepped since 2001. Then the NRC began an alternative approach to compliance based upon an

industry lobbyist standard NFPA 805. The premise of the new approach lobbied by NEI and the NFPA is currently being codified by direct reference of NFPA 805 into federal regulations. It is based solely on probabilistic analysis, improperly grounded in unsubstantiated assumptions regarding fire event probabilities.

The Energy Policy Act of 2005 (EPAct) , in response to September 11, 2001, compelled the NRC to improve fire protection coping ability across the nations fleet, yet instead of improving fire protection, the NRC is systematically reducing fire safety measures.

HISTORY OF FIRE SAFETY ISSUES

1993 – Congress Together With The NRC Office Of Inspector General Responded To Symptoms indicating a Troubled Agency:

In 1993 Congress called for hearings on Fire Protection, to correct problems with a fire-retarding material at nuclear power plants. The Justice Department began a criminal investigation into whether the NRC and the nuclear industry were misled about the fire-retarding capabilities of Thermo-Lag, a gypsum-like material used to protect critical electrical wires at nuclear power plants in case of fire in 1993. See Exhibit FP No. 1

Under NRC regulations, the retardant material must be able to withstand very high fire temperatures -- for one hour if the plant has a sprinkler system, three hours if it doesn't. The current situation with HemyC, unfortunately is reminiscent of Thermo-Lag.

Investigations found Thermo-Lag was approved as a protective barrier in the early 1980s. The NRC staff, however, never conducted independent tests to determine if the material met federal standards.

According to Leo Norton, the NRC's Assistant Inspector General of Investigations, in one test, **THERMO-LAG collapsed within 22 minutes**. He also said the NRC never bothered to personally test the product, preferring to take the word of vendors and utility company officials who swore under oath test results showed the product worked.

The Office of the Inspector General said NRC staff members who approved the fire-protective material "operated under the premise that the information was accurate because it was submitted under oath." The material in question, Thermo-Lag, was used in 79 nuclear power plants nationwide. See exhibit FP No. 2

During a 10 year period there also were a number of reports - some from utilities - indicating that the material failed to meet NRC requirements,

including one that it produced toxic gases when burned. But each time, the NRC failed to pursue them, agency investigators said.

David Williams, Inspector General for the U.S. Nuclear Regulatory Commission, also told lawmakers the NRC " that, "Between 1981 and 1991, the NRC staff did not observe any tests of THERMO-LAG. Further, the NRC staff did not investigate the qualifications of or visit the laboratory which purportedly supervised most of the THERMO-LAG tests."

"The NRC blindly accepted the utilities' assurances," said Rep. John Dingell, D-Mich., chairman of the subcommittee and of the full Energy and Commerce Committee. "This is hardly a regulatory success." He charged that the use of THERMO-LAG has resulted in "substandard fire protection" for nuclear plants that employ the material.

In response to these allegations, nuclear power plant officials said they're taking added safety precautions, some of which have been ordered recently by the NRC.

NRC "inquiries to date indicate that repairs of upgrading may be needed," Selin said the agency is holding off on further action until it has "adequately identified what criteria are appropriate to decide what standards have been met." See Exhibit FP No 3.

Implementing Risk-Informed, Performance-Based Fire Protection

The Commission approved the 50.48(c) rule in May 2004, and published the rule in June. It took effect in July.

The Commission also unlawfully allowed the staff to use its discretion in enforcing certain fire protection issues for plants transitioning to the new rule. The enforcement discretion provided an incentive for licensees to adopt NFPA 805, even though it is completely unlawful.

It provided a "get out of jail card" for non-compliant licensees that failed to implement the rules enacted in 1979 with no penalty for violating federal rules and risking the health and safety of the public for decades. Subsequently, by the end of February 2006, operators of 42 reactors had sent letters of intent indicating their commitment to adopt the voluntary standard.

Manual Fire Safety Protection

Licensees are required to protect plant equipment necessary for safe shutdown using a combination of physical separation, barriers, and methods to detect and control or extinguish fires. The NRC has also reviewed and approved operator manual actions, as another acceptable method, to safely

shut down the plant in the event of a fire. An example is manually opening a valve to prevent it from closing improperly during a fire.

There are a substantial number of licensees relying on operator manual actions that have not been reviewed and approved by the NRC to mitigate fires in fire areas with redundant safety trains (commonly referred to as III.G.2 areas since Section III.G.2 of Appendix R to 10 CFR 50 provides the requirements).

The NRC staff proposed a rule change that would enable the licensee to demonstrate acceptability of manual actions used to safely shut down a plant in the event of a fire. **The rule's primary objective was to improve efficiency by minimizing the number of exemption requests.** This is an unacceptable rationale for avoiding the basis of federal rules enacted in 1979, in response to the Browns Ferry fire.

Stakeholders contend that the current failure of fire protection at Indian Point and the NRC rushed approval of the amended exemption request that reduces the 1 hour requirement to only 24 minutes is a violation of the Presidential Order to protect nuclear power plants against Design Basis Threats- partially codified in 10CFR73.1.

In defiance of Congress, the NRC has stripped down the rules by using so called "alternative analysis" favored by the nuclear industry and the nuclear industry lobbyists. "Alternative analysis" is a cost benefit analysis disguised as a probabilistic analysis being codified in 10CFR50.48(c) . Profits of the nuclear industry are now being weighed against protection of public health and safety. Unfortunately it appears that the bias is leaning heavily in favor of corporate profits.

Stakeholders contend that the NRC has wrongfully granted the exemption from fire safety regulations for the following reasons of fact and law, that are within scope of the license amendment.

1. 24 minute exemption to a Appendix R, and 10CFR50.48 are incorporated into the plants operating license, and is as a matter of fact and law, an amendment to the operating license.

2. Fire or fires could be set by insiders, and could quickly bring down both Indian Point 2 and Indian Point 3, based on the 24 minutes rule, in violation to the Design Basis Threat10 CFR 73.1.

3. A fire caused by an aircraft penetrating a two foot thick above ground tunnel could not be extinguished in 24 minutes and could prevent safe shut down.

4. The original exemption request July 24, 2006, was for a reduction from 1 hour to 30 minutes. Then after the license renewal application has already been submitted by Entergy, Entergy amended the exemption request from 30 minutes to 24 minutes. See exhibits FP No, 5 and Exhibit FP No. 6

The public was not aware of this. Although the NRC could not have done an adequate independent Safety Evaluation in a few weeks, the NRC approved this in a only nine weeks later.

NRC staff have explained that the NRC approved the exemption on the bet that the industry would fully adopt NFPA 805, Performance based Standard for light water Reactor Electric Generating Plants, 2001 edition, now codified under 10CFR50.48(c).

5. The NRC is aware of multiple plants directly defying the present rules regarding fire protection with prima facie evidence in operational procedures of depending on manual actions to save essential equipment, without exemptions even requested. The NRC approved the amended exemption request in violation of promises to Congress to correct deficiencies from a similar material failure – thermolag affecting 79 plants—instead tolerating of deficiencies.

6. The exemption was argued by Entergy as not requiring an environmental assessment—because the previous exemptions did not require the assessment. This again is a fatally flawed argument, the difference between fire protection of 1 hour instead of 24 minutes has significant Environmental consequences, that must be fully understood. The NRC approval of this exemption is a violation of NEPA.

7. The NRC has violated §51.101(b) in allowing changes to the operating license be done concurrently with the renewal proceedings. The exemption request was

modified by Entergy on August 16, 2007 for IP3, only two weeks after of the License Renewal Application Renewal was accepted by the NRC on August 2, 2007. The exemption was then approved and published on October 4, 2007, without public involvement and in defiance of §51.101(b).

Therefore Stakeholders contend that NRC wrongfully granted Entergy the amended exemption request, filed in the Federal Registry on October 4, 2007, thereby reducing adequate protection of public health and safety, by reducing the fire safety requirement from one hour to 24 minutes.

CONTENTION 5: The Fire Protection Program described in the Current License Basis Documents including the unlawfully approved exemptions to Appendix R, the Safety Evaluation and the amended license for Indian Point 3 fail to adequately protect the health and safety of the public, and fail to meet the requirements of 10 CFR 50 and Appendix R

ISSUE: Allowance of conditions that require a fire to be extinguished in the unreasonably short time span of 24 minutes or else risk a complete loss of control of crucial safety systems is unacceptable and significantly increases

the likelihood of uncontrolled reactor criticality, inadequate cooling of the reactor core and the potentially catastrophic outcome of a core melt.

Background and Summary of Contention

The fire protection program advanced by Entergy for IP 3 is deficient in that it fails to safeguard the control room operation of achieving safe shutdown of the reactor in the event of a significant fire. The program is based on preposterously optimistic time and capability assumptions that significantly increase the likelihood of uncontrolled reactor criticality, inadequate cooling of the reactor core and the potentially catastrophic outcome of a core melt.

Specifically, the highly implausible scenario upon which Entergy gambles is that: fire ignition, fire detection, confirmation thereof, a determination of proper control acts, fire brigade formation and dispatch, and conflagration extinguishment, can all occur in a time span of less than 24 minutes. Moreover, under conditions of high heat, choking and blinding smoke and with electrically energized circuits present, plant responders will also be able to save operability of major cables required for safe shutdown. And all of the necessary actions and outcomes may be relied upon, even should the fire be one of several unfolding plant emergency conditions.

Entergy's dubious fire protection plan is part and parcel of a series of requests for exemptions from critical and long-standing fire (and other) safety regulations. The basic fire safety regulatory scheme was instituted nearly 30 years ago after a major fire at the Browns Ferry nuclear plant in Alabama, burned out of control for almost seven hours and nearly disabled the reactor's emergency core cooling system.

To reduce the critical threat, exposed by Browns Ferry, of a fire disabling all redundant safe shutdown electrical circuits in the same zone of a nuclear power plant, regulations were enacted to require either significant physical separation between cable trays and conduits, or the use of physical fire barriers. Fire barriers can be in the form of fireproofing material or insulation wraps. However the barrier must be qualified to withstand standardized American Standard Test and Measures (ASTM) E-119 furnace conditions. [Section III.G. of 10 CFR 50 Appendix R.]

At IP3, one such fire barrier employed is an insulation system known under the brand name HemyC, which is required to be able to withstand fire conditions for at least 1 hour (as per the requirements of 10 CFR 50.48,

Appendix A, Branch Technical Position 9.5.1, and Appendix R). The 1 hour period was designated as necessary to protect safe shutdown power, instrumentation and control circuits from fire damage in the event of a significant fire.

In 2005, however, independent laboratory tests revealed that Hemcy, could, in fact, fail in as little as 15 minutes. According to published test results, the HemyC material was identified to shrink under standardized fire test conditions, opening seams and exposing electrical circuits vital to the safe shutdown of the reactor to fire damage, potentially rendering them inoperable as well as introducing electrical short circuits to safety significant associated circuits.

In response to this safety problem, Entergy has asked the NRC for an exemption from the rule requiring the fire barrier to be able to hold up for at least 1 hour. In doing so, Entergy has effectively asked the NRC to alter the very assumptions of how a fire can affect areas containing critical plant cabling and equipment and how long fires might last.

Simply put, Entergy wants the NRC to degrade the fire safety rules to accommodate Indian Point's degraded fire safety condition.

A Viable Protection Program is Central to the Safety of a Nuclear Power Plant

The NRC "Severe Accidents study (NUREG-1150) recognized that fire is a significant risk contributor to core damage frequency, as much as 50 percent of the total risk and that fire can both initiate a nuclear accident and compromise the operator's ability to control reactor shutdown and maintain it in stable cool down. This study further recognized that a typical nuclear power station will have 3 to 4 significant fires.

As a preliminary matter, a fire protection program must take due cognizance of the realities of fire. (This should be obvious, but the posture of Entergy indicates that such realities are not apparent to all.)

The Applicant requested the NRC grant an exemption from federal rules for a extinguishing a fire in the tunnel whose duration was unknown. Applicant stated that class 1E cables in trains separated by less than 12 inches would be inoperable in less than 24 minutes. These cables are vital for operating both normal and emergency systems for the safe operation and emergency shutdown of the plant.

Loss of these power cables together with diminished operation of safety related valves, (such as, Pressurizer Operator Relief Valve, Core Spray System operation, or the Charging System), which may reasonably be anticipated during a tunnel fire, can render the reactor energy uncontrolled and the reactor condition degradation immitigable. Both control and Power cables run through the two tunnels. See exhibit FP No. 9, and 10. On December 17, 2003, President Bush issued Homeland Security Presidential Directive 7 (HSPD-7), which supersedes portions of PDD-63 and clarifies that the Department of Energy is the lead agency with which the energy industry will coordinate responses to energy emergencies.

This condition has been known since 1995, See exhibit FP No. 8 when NRC inspectors reviewed the in-progress plans to install an untested fire wrap HemyC in the tunnels, and acknowledged lack of ASTM 119 testing. Despite these issues, the NRC inspectors approved the modification with the understanding that testing of the wrap would be done at a later date. Doing this allows Applicant to, in effect, make "an agreement to agree".

It defies logic that 11 years, later the NRC declared the HemyC material unacceptable to meet 1 hour fire limits when it published Generic Letter 2006-03.

The improper design of the tunnel and the susceptibility of the tunnel to single failure criteria was identified in 1976, in a report by the Project Manager, Division of Project management, U.S. Nuclear Regulatory Commission on February 6, 1976. As early as this report, the operator and the NRC both knew that both tunnels were required to be functional in order to safely shut the plant down. . See page 19 of Exhibit FP no. 10 where the NRC points out that system logic requires that two, of out three, systems be operable following an accident.

In addition, the problem of associated circuits was not dealt with at all. This entire issue languished for years. The 1995 NRC inspection report acknowledges use of HemyC material inside containment. Yet, the Applicant's LRA does not provide a resolution of unacceptable burn times for that configuration.

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, [Section] 50.48, requires that nuclear power plants that were licensed before January 1, 1979, including IP2 and IP3, must satisfy the requirements of 10 CFR Part 50, Appendix R, Section III.G. Subsection III.G.2 addressing fire protection features for ensuring that one of the redundant trains necessary to achieve and maintain hot shutdown conditions remains free of fire damage in the event of a fire. Subsection III.G.2.c provides use of a 1-hour fire

barrier, fire detection and automatic fire suppression in the area, as a method to comply with this fire protection requirement.

In an NRC letter and safety evaluation (SE) dated February 2, 1984, the NRC improperly granted the applicant exemptions from the requirements of Appendix R, Section III.G.2, for Fire Area ETN-4 (Fire Zones 7A, 60A and 73A). The exemption was applicable where redundant safe-shutdown trains are not separated by more than 20 feet, without intervening combustibles or fire hazards, and that redundant safe-shutdown trains are not separated by 1-hour rated fire barrier in an area protected by automatic fire detection, and suppression systems.

The exemption was based on the minimum of 12" spatial separation between the redundant trains, minimal fire hazards in the area, the use of asbestos-jacketed flame-retardant cables, and the installed automatic fire detection and cable tray suppression systems.

Following a comprehensive reassessment of the IP2 & IP3 Appendix R compliance basis, the need for additional separation measures was identified and the untested fire barrier system was installed to provide 1-hour rated fire barriers on several redundant safe-shutdown raceways in Fire Area ETN-4 (Fire Zones 7A, 60A and 73A) for Unit 3. By Safety Evaluation dated January 7, 1987, the NRC accepted the use of 1-hour rated

fire barriers in the above fire area and confirmed continued validity of the exemption granted by the February 2, 1984 SE. IP3 used the untested HemyC fire barrier system to provide the 1-hour rated fire barriers. In the January 7, 1987 SE, the NRC also approved an exemption from Appendix R, Section III.G.2, separation requirements for Fire Area PAB-2 (Fire Zone 1) allowing redundant safe-shutdown trains to be separated by more than 20 feet without intervening combustibles or fire hazards, and with an automatic suppression system. [redacted]

This exemption required physical separation between redundant safe shutdown trains; low fire loading in the area; and continuation of the existing fire protection features, including an automatic fire detection system, manual hose stations and portable extinguishers; a partial-height non-combustible barrier designed to protect redundant equipment against radiant heat from a fire; and a 1 hour rated HemyC cable wrap around the normal power feed to the redundant Component Cooling Water (CCW) Pump 33.

Testing by a laboratory retained by the NRC in 2005 identified HemyC electrical raceway fire barrier system (ERFBS) as a nonconforming barrier, potentially failing in a little as 13 minutes and thus, not capable of providing a 1-hour fire rating, and Information Notice (IN) 2005-07,

"Results of HEMYC Electrical Raceway Fire Barrier System Full Scale Fire Testing," Exhibit FP no. 11 and Generic Letter (GL) 2006-03, "Potentially Nonconforming HemyC and MT Fire Barrier Configurations," were issued to licensees to inform them of the issue and to collect information regarding HemyC fire barrier installations.

In response to GL 2006-03, the Applicant informed the NRC that it declared the HemyC Electrical Raceway Fire Barrier System Full Scale Fire Testing RFBS, IP3 inoperable, and implemented temporary compensatory measures, including an hourly fire watch and verification that fire detection systems are operable in the affected fire areas until compliance is restored for the HEMYC Electrical Raceway Fire Barrier System Full Scale Fire Testing.

In a letter dated July 24, 2006, Applicant stated it would modify the installed HemyC ERFBS to provide only a 24 minute rated fire barrier for cable tray configurations and a 30 minute rating for conduit and junction box configurations between redundant trains of safe shutdown equipment and cables, i.e., allowing for fire barrier failure in less than half the time as the previously approved 1-hour fire barrier. Applicant asserted that IP3 did not need to employ a 1 hour fire barrier because there were minimal fire hazards and fire protection features in the affected areas.

In summary, by letter dated July 24, 2006, and supplemental letters dated April 30, May 23, and August 16, 2007, Applicant requested revisions to the pending exemptions from fire safety regulations for the Upper and Lower Electrical Tunnels (Fire Area ETN-4, Fire Zones 7A and 60A, respectively) and the Upper Penetration Area (Fire Area ETN-4, Fire Zone 73A), to allow only 24 minute rated fire barriers be used to protect redundant safe shutdown trains in lieu of 1 hour rated fire barriers. For the 41" Elevation CCW Pump Area (Fire Area PAB-2, Fire Zone 1). Applicant requested the existing exemptions to be revised to allow for only a 30 minute rated fire barrier to protect redundant safe shutdown trains located in the same fire area.

Besides the obvious reduction in adequate protection to public health and safety, the blinding speed that this exemption was granted, is stunning. It is doubtful that the NRC staff was able to rigorously evaluate the significant change in only a few short weeks.

Furthermore, this reduction allows fire protection at nuclear power plant sited within 50 miles of over 20 million people, to be inferior to that required by New York State Building codes, which require a provide either 1 or 2 hour firewalls in commercial buildings, depending on use.

There are numerous sufficient alternatives that could be used to retrofit the plant, to restore fire protection to at least one hour. This exemption is clearly a reduction of safety rules made to accommodate the financial interest of the Applicant, and is clear violation of the NRC's mandate to protect public health and safety.

Discussion

Pursuant to 10 CFR 50.12, the NRC may grant exemptions from the requirements of 10 CFR Part 50 when:

> (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present.

One of these special circumstances, described in 10 CFR 50.12(a)(2)(ii), is that the application of the regulation is not necessary to achieve the underlying purpose of the rule. The underlying purpose of Subsection III.G.2 of 10 CFR 50, Appendix R, is to ensure that one of the redundant trains necessary to achieve and maintain hot shutdown conditions remains free of fire damage, in the event of a fire. The provisions of III.G.2.c through the use of a 1-hour fire barrier with fire detectors and an automatic fire suppression system is one acceptable way to comply with this fire protection requirement.

However, Applicant's most recent amendment to the exemption, modified it to reduce the requirement to 24 minutes was dated August 16, 2007. This was a modification of their exemption request dated July 24, 2006 in which they requested a reduction of the 1hour minimum requirement to 30 minutes. In addition on August 16, 2007 the Applicant acknowledged that in order to meet the reduced time of 24 minutes, it would require a modifications.

This is a significant amendment of IP3's operating license, as allows for far less than the minimum of 1 hour, fails to provide adequate protection and lacks even the most basic foundational support. Such an analysis, for example, would patently require a detailed description of modifications that would need to be made to the cable trays and junction boxes in the tunnel.

Stakeholders strongly object to the exemption being granted. The scenario upon which Entergy gambles, to wit: fire ignition, detection, confirmation, determination of proper control acts, fire brigade formation and dispatch, and extinguishment – all in less than 24 minutes – under conditions of high heat, smoke and with electrically energized circuits present, is profoundly implausible. Significantly, Applicant proffers no evidence that this scenario has been adequately tested or can be relied upon.

Indeed the broadly available literature on fire safety as well as plain common sense leads to the conclusion that placing confidence in Applicant's scenario is foolhardy.

The Applicant asserts that fire hazards and ignition sources in both Fire Areas ETN-4 and PAB-2 remain materially unchanged from those described in the Safety Evaluations dated February 2, 1984, and January 7, 1987. For Fire Area ETN-4, the ignition sources consist of limited transient combustibles (in all fire zones), and several instrument cabinets and a 3kVA 480V/120V instrument power transformer in Fire Zone 73A.

Significantly, the class 1E cables in trains, separated by less than 12 inches, could well be rendered inoperable in under 24 minutes. These cables are vital for operating both safe operation and the emergency shutdown of the plant. Degradation or destruction of these power cables together with loss of full operation of safety related valves (such as the Pressurizer Operator Relief Valve, the Core Spray System or the Charging System) would be reasonably likely to occur during a plant fire in this tunnel. Under such circumstances, the 30,000 BTU of reactor energy could be rendered uncontrolled and the reactor condition degradation would probably be immitigable.

Stakeholders assert the following: (1) the fire hazards analysis and the fire safe shutdown analysis are living documents that are an element of the Current License Basis. These documents require examination and reanalysis as the Applicant implements modifications to the facility. (2) The 1984 analysis was not updated until well beyond 10 years. The most recent safe shutdown analysis appears to be revision 2, dated August 2000, which is more than seven years out-of-date. Thus these analyses are historical and void, given the reality that modifications were made to the facility during the intervening years. Without the baseline analysis being kept current, it is essentially impossible for engineering analyses, engineering design changes, operational function changes and even the most fundamental changes to the facility, to be performed in conformance with 10 CFR 50.48 and Appendix R.

The 24 minute minimum can only be obtained after modifications of the cable trays and boxes occurred, such modification many not even be made until 2008. Thereby leaving the current unsafe conditions of non-compliance with Appendix R.

For the 41" Elevation CCW Pump Area (PAB-2, Fire Zone 1), the current IP3 Fire Hazard Analysis indicated a fire severity of less than 10

minutes. Combustibles include the CCW pump bearing lubricating oil and transient materials.

The HemyC-wrapped Box-Type Configuration installed in Fire Area ETN-4 (Fire Zone 73A) is comparable to Configuration 2G in NRC Test 2, *except for the lack of the stainless steel over-banding*. These enclosures are protected by a direct-attached 2"-thick HemyC blanket wrap. Both NRC and industry-sponsored tests of fire protection cable function when tested in accordance with ASTM E-119. To more closely reflect Configuration 2G, the Applicant is committed to install over- banding on the Box-Type Configuration at IP3. Cable Tray Configuration The HemyC-wrapped Cable Tray Configuration installed in Fire Area ETN-4 (Fire Zones 7A and 73A) is comparable to Configuration 2B and 2D of NRC Test 2. These cable trays are protected by a 1-1/2"-thick HemyC blanket wrap with a nominal 2" air gap between the protected cable tray and the blanket.

Fire tests conducted by both NRC and industry indicated that these HemyC-wrapped cable tray configurations will provide up to 24 minutes of thermal protection in accordance with the ASTM E-119 time-temperature profile.

The Applicant stated that administrative controls of hot work and transient combustibles allowed designated Fire Areas ETN-4 and PAB-2 as

"Level 2" combustible control areas, which constrain transient combustibles to "moderate" quantities as follows, in both IP2 and IP3:

- 100 pounds of fire retardant treated lumber, or
- 25 pounds of loose ordinary combustibles or plastics, or
- 5 gallons of combustible liquids stored in approved containers, or
- One pint of flammable liquids stored in approved containers, or
- One 20 ounce flammable aerosol can.

With the proposed additional protection of electrical raceway supports and installation of over-banding on HemyC box configurations, the modified fire barrier configurations are expected to afford at least 24 minutes for cable tray configurations and 30 minutes of protection for conduit and box configurations; 50% or less than the time required by Design Basis.

Since the HemyC electrical raceway fire barrier system is expected to provide protection for redundant components and cables in the event of a fire, the NRC staff, inappropriately, concluded that the minimal combustibles in the areas and existing active/passive fire protection features can compensate for the reduction in Defense-in- Depth of objectives 3 and would not impact IP3 post-fire safe-shutdown capability.

Stakeholders disagree with this conclusion. Material facts in genuine dispute include the following:

(1) The proffered findings are not demonstrably applicable to IP3. Namely, the use of HemyC wrap to protect cabling critical for control and safe shutdown of the plant is based solely upon generic testing. No test configuration matches the conditions of the HemyC wrapped cable in the IP3 tunnel. Applicant is thus engaging in unsubstantiated speculation regarding longevity of the cable function.

(2) The unique characteristics of the EDG output voltage of 480 volts (as compared to 4160 volts) impose a much higher amperage through the cables, necessitating larger gauge cable and more energy lost in power transmission in the form of heat. The tested configurations do not account for these conditions, which are unique to Indian Point's emergency generators, and buses.

(3) The scenario upon which Entergy gambles, to wit: fire ignition, fire detection, confirmation, determination of proper control acts, fire brigade formation and dispatch, and extinguishment – all in less than 24 minutes – under conditions of high heat, smoke and with electrically energized circuits present, is highly unlikely, and

cannot be relied upon as credible. Notably, in addition to putting out the blaze, plant responders would also need to save operability of on train and major cables required for safe shutdown.

Expert opinion by Ulrich Witte as former Project Engineer for the Appendix R Program to the Sacramento Utilities District Rancho Seco plant is provided in his Declaration contained in Exhibit FP-7.

Inadequate Justification for Invoking 10 CFR 50.12

The exemption the Applicant has sought would allow use of a fire barrier expected to provide less than 1 hour of fire protection. Stakeholders assert that the grant of this exemption constitutes an abuse of the Commission's discretion and violates the letter and spirit of the Atomic Energy Act of 1954, as amended.

These regulations, §10 CFR 50.12 and Appendix R were promulgated specifically in response to the 1975 Browns Ferry accident.

Brown Ferry continues to provide a particularly dramatic example of how quickly a nuclear plant can be put in jeopardy and how difficult responsive action can be. The Browns Ferry fire burned out of control for some 7 hours with temperatures as high as 1500 degrees Fahrenheit. Within

15 minutes of initiation, a high number of safety-related circuits were destroyed. By the time it was extinguished, 1600 electrical cables, including 628 safety-related circuits needed to shut down the reactor and keep it cool, coolant had been destroyed. In a 1976 report prepared by the Union of Concerned Scientists, entitled "Browns Ferry: The Regulatory Failure," the investigators noted that thick smoke, the chaos resulting from the loss of control over equipment, and inadequate breathing apparatuses made it difficult for operators to save the plant. The report revealed that the operator's nuclear engineers had stated privately to the investigators "that a potentially catastrophic radiation release from Browns Ferry was avoided by 'sheer luck.'"

Twenty million residents living within 50 miles of Indian Point Units 2 & 3 should not have to depend on "sheer luck". The NRC has the responsibility to maintain reasonable regulations with regard to fire safety protection that will adequately protect public health and safety. Stakeholders assert that a grant of Applicant's request for exemption would abuse the authority granted to the NRC by Congress.

The underlying purpose of Subsection III.G.2 of 10 CFR Part 50, Appendix R, is to ensure that one of the redundant trains necessary to achieve and maintain hot shutdown conditions remains free of fire damage

in the event of a fire. This safety margin is an imperative to protect public health and safety. It dramatically reduces the defense-in-depth criteria.

Special Circumstances: One of the special circumstances, described in 10 CFR 50.12(a)(2)(ii), is that the application of the regulation is necessary to achieve the underlying purpose of the rule. The underlying purpose of Subsection III.G.2 of 10 CFR Part 50, Appendix R, is to ensure that one of the redundant trains, necessary to achieve and maintain hot shutdown conditions remains free of fire damage in the event of a fire. As shown, this is not possible given the physical characteristics, including the layout of the cabling in the tunnel. and the material used as insulation.

Based upon consideration of the information in the Applicant's Fire Hazards Analysis, administrative controls for transient combustibles and ignition sources, previously-granted exemptions for this fire zone, and the considerations noted above, it is incorrect for the NRC staff to conclude that the Applicant's exemption request meets the underlying purpose of the rule.

There are numerous options available that do not require unacceptable risks to be placed on the safe operation, and emergency shutdown of Indian Point 2 and Indian Point3, as well as, and protection of the health and safety of the public are available.

There are no special circumstance is present, which would justify allowance the exemption requested by Entergy.

Conclusion

Stakeholders assert that Applicant and the NRC have improperly determined that pursuant the Exemption is authorized by law. The exemption is not authorized by law, as it causes an undue risk to the public health and safety and thwarts the very purpose of the regulation.

CONTENTION 6: Fire Protection Design Basis Threat. The Applicant's License Renewal Application fails to meet the requirements of 10 CFR54.4 "Scope," and fails to implement the requirements of the Energy Policy Act of 2005.

Issue Summary: Congress imposed upon the Nuclear Regulatory Commission rulemaking requirements to implement defenses against twelve distinct threats as contained under a classified documents. The Commission partially codified the Energy Policy Act of 2005 (EPAct) requirements most recently on April 18, 2007, under 10 CFR73.1, 21, 55, 56, and 10 CFR26. This contention raises issues of conformance with the *existing rule*, regardless of the controversy associated with whether the current rule fully implements the Energy Policy Act of 2005 (EPAct).

The Stakeholders assert that the existing rules as currently promulgated is within scope of the license renewal application submitted by Entergy. Yet they are not addressed in the LRA with regard to the Fire Protection Program enhancements necessary for implementation.

In fact, the Applicant has requested and has been granted an exemption to specific federal rules, that actually erodes safety at Indian Point 2 and 3, and increases vulnerability to the facility to a design basis threat that was required to be strengthened by Energy Policy Act of 2005 (EPAct).

The Applicant's LRA fails to comply with applicable law with respect to fire protection. Fire protection is one of the twelve specific components within the DBT rule. This exemption affects the current operating license, and will be carried over into the proposed new superceding license.

The Final Rule Regarding Design Basis Threat and Fire:

Congress also recognized the need for the NRC to conduct a rulemaking to update the DBT regulation in light of the events of September 11, 2001. On August 8, 2005, the President signed into law the Energy Policy Act of 2005, Pub. L. No.109-58, 119 Stat. 594, which mandated that, within 90 days, the NRC "initiate a rulemaking proceeding, including

notice and opportunity for public comment, to be completed not later than 18 months after that date, to revise the design basis threats of the Commission." *Id.* § 651, *codified at* 42 U.S.C. § 2210e. The Act specifically listed 12 factors that the NRC had to consider in conducting its rulemaking, including "the events of September 11, 2001," "the potential for attack on facilities by multiple coordinated teams of a large number of individuals," and "the potential for water-based and air-based threats." 42 U.S.C. § 2210e(b).

The NRC published its final rule in the Federal Register on March 19, 2007. 72 FR 12705 (ER 1). Although the Commission made some changes in the language of the proposed rule (adding, for example, a provision requiring defense against the threat of cyber-attacks), the agency made no changes in response to comments that had challenged its refusal to conduct an EIS, its failure to require a defense against attacking forces as large as those assembled by al Qaeda on 9/11, and against the threat of suicide attacks by large aircraft. Indeed, the Commission explicitly declined to require a defense against a force as large as that involved in the 9/11 attacks (72 FR at 12708), and it refused to incorporate any provisions concerning air attacks in the DBT (*id.* at 127 10-1 1).

1. Commission's "Reasonableness" Limit on the Design Basis Threat and the Size of the Attacking Force

Throughout the final rule, the Commission emphasized that a fundamental principle animating the DBT was that it would require a licensee to do no more than defend against attacks that a private security force could reasonably be expected to counter. As the agency put it, "The Commission has determined that the DBTs, as articulated in the rule, are based on adversary characteristics against which a private security force can reasonably be expected to defend." 72 FR at 12713.

The agency provided only one example of what might make it "unreasonable" to expect a private security force to respond to a threat: that there are "legal limitations" on the types of weapons and defensive systems available to private security forces. "Thus," the agency asserted, "it would be unreasonable to establish a DBT that could only be defended against with weapons unavailable to private security forces." *Id.* at 12714.

The NRC did not preclude the potential deliberate use of transient combustibles already available on site, to be use serendipitously to interfere with the safe operation of the facility. In fact, the rule provides that the licensee must assume that the assailant has knowledge of specific target selection and access to transient combustibles. As directed by the

Energy Policy Act, the final rule has the principal objective of making the security requirements imposed by the April 29, 2003, DBT orders generically applicable. Although specific details of the revised DBT were not released to the public, in general the final rule:

- clarifies that physical protection systems are required to protect against diversion and theft of fissile material;
- expands the assumed capabilities of adversaries to operate as one or more teams and attack from multiple entry points;
- assumes that adversaries are willing to kill or be killed and are knowledgeable about specific target selection;
- expands the scope of vehicles that licensees must defend against to include water vehicles and land vehicles beyond four-wheel-drive type;
- revises the threat posed by an insider to be more flexible in scope; and
- adds a new mode of attack from adversaries coordinating a vehicle bomb assault with another external assault.

The above reflect the need to enhance the facility against the threat of fire. However, in Entergy's most recent request for an exemption dated August 16, 2007, reducing the one hour rule contained in Appendix R to 10 CFR50 to and unacceptable 24 minutes.

The scenario upon which Entergy gambles, to wit: fire ignition, fire detection, confirmation, determination of proper control acts, fire brigade formation and dispatch, and extinguishment – all in less than 24 minutes – under conditions of high heat, smoke and with electrically energized circuits present, is highly unlikely, and cannot be relied upon as credible. Notably, in addition to putting out the blaze, plant responders would also need to save operability of on train and major cables required for safe shutdown. Under requirements of 10 CFR73.1, a single event, fire in one of the two tunnels, Fire Area ETN-4 (Fire Zones 7A, 60A and 73A) if not extinguished in less than 24 minutes violates safety margins.

CONTENTION #7: Fire initiated by a light airplane strike risks penetrating vulnerable structures.

Stakeholders contend that fire initiated by a crash or deliberate strike of an airplane crash at the facility can initiate a fire or serve fires that spread and disable critical safety systems, specifically the above ground cable tunnels. These tunnels are constructed above ground and consist of two foot concrete walls, which are easily breached by a large or even a small aircraft.

Due to the decrease in fire protection standards, and accidental or planned crash into these structures would probably cause a fire or fires,

that could not be extinguished with in 24 minutes, thereby cause safe shut down systems to become inoperable, and creating a core melt scenario.

. The reports by the Project on Government Oversight (POGO), on December, 2003 (Exhibit FP No, 12), the August 9, 2005, CRCS report to Congress by Carl Behrens and Mark Holt, Nuclear Power Plants: Vulnerability to Terrorist Attack Exhibit FP No, 13 and the Council on Intelligent Energy & Conservation Policy (CIECP) Comments to Proposed Rule 10 CFR 50,72, and 73 regarding Power Reactor Security Requirements at License Nuclear Facilities, filed with the NRC on March 27, 2007 Exhibit FP no. 14 are referred and fully incorporated, as if set forth herein.

In a 2005 updated, report by Carl Behrens and Mark Holt, Nuclear Power Plants: Vulnerability to Terrorist Attack Exhibit FP no. 15 "Protection of nuclear power plants from land-based assaults, deliberate aircraft crashes, and other terrorist acts has been a heightened national priority since the attacks of September 11, 2001. the industry has been too slow and that further measures are needed.

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.

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-demonstrate that the NRC considers such attacks to be reasonably foreseeable for purposes of requiring a NEPA review.

There is no no-fly zone over Indian Point. This presents a clear and significant danger since planes of all shapes and size, including private jets and large commercial planes. There are at least 7 major airports within the 50 miles of Indian Point, including Westchester County Airport, Stewart International Airport, JFK International Airport, La Guardia Airport, and Newark International.

International carriers are planning to use the plane for flights in and out of Kennedy,. In January 2008, Airbus will be flying into Steward Airport, located approximately 9 miles from Indian Point. Airbus's super jumbo A380, the world's largest passenger plane, It has a wingspan almost as long as a football field, it is

eight stories tall, and it weighs 118 tons heavier than the Boeing 747, the planes that were used in the terrorist attack on 9/11. "The biggest purchases of Airbus are from the United Arab Emirates., the craft is certified to carry up to 853 — about twice the capacity of the biggest version of the Boeing 747". (March 2007 NYT).

The residents in the Hudson Valley and specifically Rockland County, all of which is within 20 miles of Indian Point, , have been recently advised of the FAA's decision to increase air traffic in the region by 600 flights a day.. On average every two to three minutes the noise of aircraft flying overhead will be heard, and danger from an accidental or inital crash into the vulnerable above ground part of the plant are greatly increased.

Yet the fire protection has been decreased by more than 50%, due to the NRC's improper approval of Entergy's modified Exemption Request.

<u>The Cost Rationale is flawed as found under 10CFR12</u>

The NRC "disagreed" with comments that urged it to make clear that licensees were required to defend against an attacking force *at least* as large as the 19 attackers assembled by al Qaeda on September 11, 2001. *Id.* at

12708.[1] Instead, the NRC stated that the limit on the size of the attacking forces incorporated in the DBT was based on the "reasonableness" concept. The DBT, in the NRC's words, "represents the largest adversary against which the NRC believes private security forces can reasonably be expected to defend." *Id.* at 12714.

The NRC acknowledged that consideration of costs would be unlawful. *See id.* The NRC did not, however, explain how "reasonableness" figured into a limit on the *size* of the attacking force (and hence the size of the defending force) if it was not a cost-based consideration. The Commission also denied that the reasonableness limitation was a violation of its obligation to ensure adequate protection of the public:

> "The rule text set forth at § 73.1 represents the largest adversary against which the Commission believes private security forces can reasonably be expected to defend. Thus, when the DBT rule is used by licensees to design their site specific protective strategies, the Commission is thereby provided with reasonable assurance that the public health and safety and common defense and security are adequately protected. *Id.*

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[1] These comments did not ask the Commission to say exactly how many attackers it was requiring licensees to defend against, as such a disclosure would create an obvious risk that an attacker would tailor the size of its force to exceed that specified in the rule. Rather, commenters urged the Commission to make clear that the DBT required defense against forces the size of the 9/11 attack groups, but not that it was limited to groups of that size, but its explanation on this point amounted only to the assertion that adequate protection of safety and health somehow followed logically from the reasonableness limit:

Elsewhere, the Commission appeared to acknowledge that the defense forces required by the DBT would not be "adequate" if attacked by a force larger than the Commission felt it was "reasonable" to expect a private security force to defend against, but it stated that it was "confident" that the defenders would still try their best if attacked by such a superior force:

> Within this requirement is the expectation that, if confronted by an adversary beyond its maximum legal capabilities, on-site security would continue to respond with a graded reduction in effectiveness. The Commission is confident that a licensee's security force would respond to any threat no matter the size or capabilities that may present itself.

Stakeholders assert that the exemptions and the failure to adequately Indian Point from the threat of a rapidly spreading fire a wholly untenable risk to public health and safety. Approval of this exemption constitutes a violation of the law and the principal mandate of the Atomic Energy Act and violates 10 CFR 73.1.

CONTENTION 8: The NRC improperly granted Entergy's modified exemption request reducing fire protection standards from 1 hour to 24 minutes while deferring necessary design modifications.

In the proposed exemption request filed on July 24, 2006, whereby Entergy requested a reduction from 1 hour to not 30 minutes was not inconsequential. But then, the amended request August 16, 2007, to less

than 24 minutes and predicated upon design modifications being implemented, is a significant change to the exemption request and a substantial reduction in fire protection.

Full-scale fire tests recently performed by the NRC revealed that HemyC, a fire barrier system used to protect cables in electrical raceways in nuclear power plants, does not perform as designed. The outer covering of the barrier can shrink during a fire, opening joints in the material and potentially allowing the fire to damage cables inside. These results show that HemyC does not serve as a fire barrier for the full hour required.

Despite these new test that identified that HemyC could not withstand a fire for more than 24 minutes in certain cable set-us, required to be 1 hour it is still be used at Indian Point 3. The NRC issued Generic Letter 2006-03 in April 2006 to ensure that the affected licensees take appropriate corrective actions.

On August 16, 2007, Entergy notified the NRC that deficient design of the HemyC fire wrap would not withstand the originally proposed exemption of 30 minutes, but for an unknown duration with a best guess of 24 minutes --- and that guessed duration would only be *after plant*

modifications were completed. The necessary modifications may remain unimplemented up to December 2008.

There was no public comment period . The changes made to the proposed exemption on August 16, 2007 where never made formally public, and *almost no one noticed* until after the grant. New York State Attorney General's Office who objected, on the same day, believed that the exemption was still pending.

Complete and proper analysis of the implications on fire safety caused by the greatly reduced fire standard usually takes months. However, in a matter of a few short weeks the amended exemption request was accepted by the NRC.

The affect of NRC's grant of the October 4, 2007 exemption, are 1) reduction of fire safety parameters by more than 50%; 2) non-compliance by the operator for more than 10 years, is now pardoned, despite long term safety violations; 3) failure to consider public comment; and most importantly, 4) erosion of the time available to detect, respond and extinguish a fire that affects both *power* of emergency core cooling systems and the *controls* for those emergency systems and for normal control of reactor criticality itself.

Stakeholders contend that the NRC improperly granted the exemption request, that in fact is an license amendment, without allowing for public comment. Therefore Stakeholder request a hearing on all the exemption request reduction to 24 minutes.

CONTENTION 9: In violation of promises made to Congress the NRC did not correct deficiencies in fire protection, and instead have reduced fire protection by relying on manual actions to save essential equipment.

In bold violation of promises to Congress to correct deficiencies from a similar material failure – thermolag affecting 79 plants, the NRC instead has accepted deficiencies in fire safety. The current approval of the exemption for Indian Point requiring manual actions to save equipment is unconscionable and fails to adequately protect public health and safety.

The NRC was aware of multiple plants directly defying the present rules regarding fire protection with prima facie evidence in operational procedures of depending on manual actions to save (not repair) essential equipment without exemptions even requested.

In 1993 Congress called for hearings on Fire Protection, to correct problems with a fire-retarding material at nuclear power plants. The Justice Department began a criminal investigation into whether the NRC and the

nuclear industry were misled about the fire-retarding capabilities of Thermo-Lag, a gypsum-like material used to protect critical electrical wires at nuclear power plants in case of fire in 1993. Exhibit FP No. 3

Under NRC regulations, the retardant material must be able to withstand very high fire temperatures -- for one hour if the plant has a sprinkler system, three hours if it doesn't. The current situation with HemyC, unfortunately is reminiscent of Thermo-Lag.

Investigations found Thermo-Lag was approved as a protective barrier in the early 1980s. The NRC staff, however, never conducted independent tests to determine if the material met federal standards.

According to Leo Norton, the NRC's Assistant Inspector General of Investigations, in one test, **THERMO-LAG collapsed within 22 minutes**. He also said the NRC never bothered to personally test the product, preferring to take the word of vendors and utility company officials who swore under oath test results showed the product worked.

The Office of the Inspector General said NRC staff members who approved the fire-protective material "operated under the premise that the information was accurate because it was submitted under oath." The material in question, Thermo-Lag, was used in 79 nuclear power plants nationwide. .

During a 10 year period there also were a number of reports - some from utilities - indicating that the material failed to meet NRC requirements, including one that it produced toxic gases when burned. But each time, the NRC failed to pursue them, agency investigators said.

David Williams, Inspector General for the U.S. Nuclear Regulatory Commission, also told lawmakers the NRC " that, "Between 1981 and 1991, the NRC staff did not observe any tests of THERMO-LAG. Further, the NRC staff did not investigate the qualifications of or visit the laboratory which purportedly supervised most of the THERMO-LAG tests."

"The NRC blindly accepted the utilities' assurances," said Rep. John Dingell, D-Mich., chairman of the subcommittee and of the full Energy and Commerce Committee. "This is hardly a regulatory success." He charged that the use of THERMO-LAG has resulted in "substandard fire protection" for nuclear plants that employ the material.

In response to these allegations, nuclear power plant officials said they're taking added safety precautions, some of which have been ordered recently by the NRC.

NRC "inquiries to date indicate that repairs of upgrading may be needed," Selin said the agency is holding off on further action until it has

"adequately identified what criteria are appropriate to decide what standards have been met."

Stakeholders assert that the issues with regard to the failure of ThermoLag to perform as advertised, put the NRC on notice to adequate perform test on other similar materials, such as HemyC. The NRC subsequently failed to properly test HemyC, used at Indian Point 3.

Stakeholders contend that NRC improperly approved Entergy amended exemption request. Stakeholders further contend that the NRC must order retrofits to bring Indian Point 3 into compliance, not reduce the standards of the regulations to meet non-compliant facilities.

CONTENTION No. 10: (Unit 2) Cable separation for Unit 2 is non-compliant, fails to meet separation criteria and fails to meet Appendix R criteria. This has been a known issue since 1976; and again in 1984, yet remains non-compliant today.

Summary of the issue: Unit 2 electrical separation was done under unapproved criteria as noted in Contentions 22-26. The consequences of this include a single electrical tunnel, housing both safety related trains with about 12 inches of separation, as well as control circuits for emergency core cooling of the reactor. This approach fundamentally violates general design criteria, and does not comply with even the draft criteria issued July 11, 1967 or with Appendix R criteria. A single event, a fire in the electrical

tunnel could sever control of the reactor from the control room and create a scenario analogous or far worse than what happened at the Browns Ferry fire of 1975.

This issue was raised in a report written in 1976, by the Nuclear Regulatory Commission, and again raised by in 1984 directly to the NRC by an employee of Unit 2 (See Exhibit FP No. 10 and Exhibit FP No. 20).

This issue relates to Appendix B, of the Applicants LRA. Essentially, the stakeholders are asserting that without adequate design measures in place that are lawful, and a reasonable assurance of the protection of the health and safety of the public, the ageing program described in the Applicants' LRA is meaningless.

CONTENTION No. 11A (Unit 2 and Unit 3): The Fire protection program as described on page B-47 of the Appendix B of the Applicant's LRA does not include fire wrap or cable insulation as part of its aging management program.

The aging management program is described includes "fire barrier penetration seals, fire barrier walls, ceilings and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained." There is no mention of fire insulation for cables themselves. See page B-47, Appendix B of the LRA. For IP2 the program

includes period testing of the Halon system, and for IP3 the aging program includes periodic testing of the CO_2 system.

Under 10CFR part 54.4 "Scope," fire barrier cable wrap and insulation clearly meet the scope definition specifically contained in §54.4(a)(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).

and,

> (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.

For clarity, this is provided below:

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

Given the numerous issues identified with cable separation design criteria inadequacy, and cable insulation design failures as described in Contentions 5-10 above, and the requirement as delineated under federal rules, the Applicants' LRA is inadequate and inaccurate, specifically regarding cable insulation and circuit separation in the event of a fire at both Indian Point 2 and Indian Point 3.. Stakeholders assert that the Applicant has not demonstrated electrical cable function for an additional 20 years, but quite frankly, it appears the Applicant has not even demonstrated compliance with current regulatory requirements.

CONTENTION 11B: Environmental Impact of an increase in risk of fire damage due to degraded cable insulation is not considered thus the Applicants' LRA is incomplete and inaccurate, and the Safety Evaluation supporting the SAMA analysis is incorrect.

Summary of Issue: The Environmental Risk derived from the SAMA analysis for IP3and IP2 is deficient because it does not include the increased risk of a fire barrier failure, and the loss of both cable trains of important safety equipment in evaluating a severe accident.

The LRA for IP3 fails to comply with the requirements of Appendix A, Criterion 3 of 10 C.F.R. Part 50 and Appendix R (Section G.2) of 10 C.F.R. Part 50 because it does not provide "enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating" nor does it meet either of the other two alternate requirements of Section G.2 of Appendix R.

This failure of compliance with fundamental safety requirements increases the risk of fire induced failure of redundant safety-related electrically operated equipment. The ER SAMA analysis does not consider the risk of electrical circuits important for safety failing to perform their function due to loss of redundant trains by fire and does not compare the costs of those larger consequences against the cost of mitigating the accident by upgrading the relevant cable and equipment enclosures to meet the requirements of Section G.2 of Appendix R.

The failure to meet fundamental design criteria is fully incorporates Contention No. 10 this petition, as fully set forth herein.

Stakeholders assert due to the failure of Applicant's LRA to comply with the requirements of Appendix A, and Appendix R, the NRC must deny the LRA.

CONTENTION 12: Entergy either does not have, or has unlawfully failed to provide the Current License Basis' (CLB) for Indian Point 2 and 3, accordingly the NRC must deny license renewal.

Statement of the issue: Stakeholders assert that the Current License Basis for Indian Point 2 and Indian Point 3 are unknown and unavailable, thereby obstructing Stakeholder' right to review and analyze plant specific commitments and modifications.

Applicant is required to have the Current License Basis (CLB) in its possession and control the precise current license basis for each unit. The CLB is defined in 10 CFR 50.3. The CLB required for license renewal under 10 CFR 2.390 is unavailable.

Rule 10 CFR 54.3 plainly states that pertinent parts of the CLB must be available at the beginning of the public review period.

Numerous attempts have been made by the NRC, as well as the General Accounting Office (GAO) to determine whether the Current License Basis is known, current, documented, and available. However, none of these efforts have been successful. The most recent was an investigation by the Office Inspector General report (Exhibit X) concluded that the CLB for each plant is *not known*. This is particularly material, given that the

pertinent parts of the CLB are required under §2.309 to be available to Stakeholders.

The CLB includes the Design Basis Document Program. For IP3 this is referred to as the Design Basis Verification Program (DVP) and for Indian Point 2, this is referred to as the Design Basis Document Program. The status of design basis programs are outdated. And the design basis documents are not reliable. See for example the Indian Point 2 & Indian Point 3 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 have been allowed to commence until Stakeholders had access to a full and complete set of the LRA and its underlying documents, including, but not limited, to the Final Safety Analysis Reports (FSARs) (all versions), USFARs (all versions), the most current and up to date operator and/or NRC version of the CLB.

Current Licensing Basis is the term of art for 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 that are docketed and

in effect. The plant specific design basis includes all modifications and additions to such commitments over the life of the license.

The Current Licensing Basis (CLB) described in 10 CRF 54.3 is:

> 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.

Stakeholders contend all referenced documents associated with the above are also part of the licensing basis and are incorporated by reference into the LRA. This includes the CLB and all granted Exemptions, Exceptions and Deviations from the CLB that will be carried over into the proposed 20 year new superseding license period.

Neither the NRC staff nor the Applicant has made the list of such grants of Exemptions, Exceptions and Deviations available to Stakeholders and interested parties, despite multiple requests. By failing to provide such a list, the NRC and Applicant are either (a) acknowledging that they do not have such a list, and therefore are not able to adequately monitor and operate the plant during the proposed 20 year new license period, or (b) inappropriately withholding pertinent information from Stakeholders.

Stakeholders contend that the NRC must deny the Applicant's LRA because the Current License Basis is required for license renewal under 10 CFR 2.336 is unavailable and unknown. In addition Stakeholders contend that the NRC must deny the Applicant's LRA because the Exemptions, Exceptions and Deviations from the CLB are either unavailable or unknown.

CONTENTION 13: The LRA is incomplete and should be dismissed, because it fails to present a Time Limiting Aging Analysis and an Adequate Aging Management Plan, and instead makes vague commitments to manage the aging of the plant at uncertain dates in the future, thereby making the LRA a meaningless and voidable "agreement to agree"

Statement of Issue: Stakeholders contend that Applicant has submitted

an LRA that contains uncertain and undefined commitments with regard to the Aging Management Plan. A new superseding license should therefore not be approved by the NRC because the LRA fails to constitute a binding agreement, and merely recites the intention of Entergy to "agree to agree" to some plan that will be delineated and put into effect at some unspecified point in the future. Entergy's vague commitment to perform aging management is, accordingly, meaningless.

Specifically, Entergy fails to provide the required Time Limiting Aging Analysis (TLAA) and adequate Aging Management Plans to deal with known plant degradation issues. Instead, it seeks to conduct certain TLAA as part of the new proposed superseding license and promises to implement, as yet unknown, Aging Management Plans at some future date and time.

The NRC's responsibility in the LRA process is to identify shortcomings and unaddressed issues in the application. Negotiating a list of future, unspecific commitments with the Applicant is not within the NRC's mandate.

Either an adequate TLAA was done to address a known aging issue, or it was not. Either an aging management plan exists or it does not. If the

TLAA was not done or an Aging Management Plan does not exist, the application is incomplete.

Moreover, the licensing process is supposed to be transparent and Entergy's LRA is supposed to provide Aging Management Plans for stakeholder and public review. The failure to provide such plans effectively removes them from public scrutiny.

The NRC should not accept an incomplete and inadequate LRA and grant a new superseding license based on Entergy's nebulous commitments to perform unspecified tasks or actions at times uncertain. Such commitments amount to nothing more than a meaningless agreement to agree.

Legal Basis: An LRA is required to be complete, and address all issues material to the proposed new superseding license. Specifically, 10 CFR 54 (A) requires a license to conduct a Time Limiting Aging Analysis (TLAA) for in scope primary equipment and components subject to fatigue, and (B) requires adequate and specific Aging Management Plans be included in the LRA to deal with in scope parts, components, and systems subject to aging issues such as fatigue.

The NRC regulations do not provide a mechanism for an Aging Management Plan to be submitted at some later date.

It is further a matter of basic hornbook law, that, to be a valid and enforceable, an agreement 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 agreement is non-binding, and is often referred to by courts as an "agreement to agree" or a statement of intent, both of which are unenforceable as contract or license. (A license is essentially a contract between a regulator and a regulated business, in this case the NRC and IP2 LLC and IP3 LLC.)

In *Richie Co. LLP vs. Lyndon Insurance Group*, Inc., a federal case out of the Eighth Circuit interpreting contract law, the Court held that the agreement at issue was not a legally enforceable agreement, but a non-binding letter of intent and agreement to agree. The Court ruled that a document creating an agreement to negotiate in good faith in the future is not enforceable, where the agreement is not the complete and final agreement governing the transaction at issue. The Court 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". The Court further found that language that

spoke of future actions and agreements contemplated but not yet completed showed that the understanding was not the complete and final governing agreement, but "merely created an agreement to negotiate in good faith." Language manifesting an intention to do something essential at a later date, thus is not binding, but merely an unenforceable agreement to agree.

Thus, in the event the NRC accepts vague commitments with unspecified protocols to be determined at an uncertain date in the future, for certain components and systems in the Aging Management Plan for Indian Point 2 & 3, then entire plan and new superceding license will be unenforceable and void.

Stakeholders further submit that a nuclear reactor Applicant should not be allowed to operate a facility without a complete and fully enforceable legal license and agreement in place prior to approval of the license.

The principle of public input and participation If the NRC approves the proposed new 20 year license based on a LRA that contains criteria and obligations of the Applicant that do not have sufficient certainty with regard to the Aging Management Plan, then essential terms and conditions of the new superseding license, that may adversely affect public health and safety, will be left vague and uncertain. Critically, allowing a future commitment

bars public Stakeholder involvement in the process, thereby removing the issue of aging management from public review.

However, since many nuclear plant licensees who have moved through the re-licensing process are finding it difficult to meet deadlines set for future commitments, the NRC is discussing the possibility of granting these licensees relief from those very commitments.

A noteworthy illustration of the illusory nature of promises is the broken promise of Indian Point to install a closed cooling system. Indian Point 2 & Indian Point 3 made promises when first licensed back in the early 70's to design and build a closed cooling system, with an originally delivery date in 1979. This deadline passes nearly 30 years ago and Entergy is now actively fighting against the installation of a closed cooling system at Indian Point.

Conclusion: Stakeholders contend that the law requires the NRC needs to act as a regulator which abides by and enforces its own rules and regulations, not as a deal maker trying to help the industry.

In the current LRA proceeding and approval process Applicant makes a commitment to the NRC to vaguely do something, left undefined, at some uncertain future date and time, after the new superceding license for 20 years

has already been issued. This amounts to nothing more than an agreement to agree.

Further, in the event the NRC accepts Applicant's LRA with uncertain and vague criteria, Stakeholders will be effectively barred from participating in the review of specific criteria that may adversely affect public health and safety. This is a violation of Stakeholders constitutional rights 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 that may adversely impact public health and safety and which would cause the proposed new superseding license to be unenforceable and void.

CONTENTION # 8 of this Petition is referenced and incorporated fully, as if set forth herein.

CONTENTION 14: 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."

Stakeholders contend that 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 disregarding the recommendation of the NRC, and even though the NRC incorporated it in next revision of License Renewal Interim Staff Guidance Supplement 1 to Regulatory Guide 4.2, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses." The Applicant failed to address not just the rule, but also failed to address the trade guidance documents as well. (see Exhibit BB).

Stakeholder content that because the LRA failed to include the Final License Renewal Interim Staff Guidance prepared on August 14, 2007 the LRA is incomplete and cannot be accepted.

CONTENTION 15: Regulations provides that in the event the NRC approves the LRA, then old license is retired, and a new superseding license will be issued, as a matter of law § 54.31. Therefore all citing criteria for a new license must be fully considered including population density, emergency plans and seismology, etc.

In January of 1989, NRC's General Counsel issued his opinions regarding extensions of operating licenses at nuclear power plants, namely

that the extension be accomplished via a **new license** rather than an amendment of the current license, and that an environmental assessment would be required with either a finding of 'no significant impact' or an environmental impact statement.

This was codified in 10 CRF 54.31 of the NRC's relicensing regulations which states:

> (c) A renewed license will become effective immediately upon its issuance, thereby superseding the operating license previously in effect.

NRC representatives including the Director of Relicensing Dr. Bo Pham, at various Stakeholder meetings confirmed, the meaning of this section, that in the event the Applicant's LRA is approved by the NRC, a NEW superseding license will replace the current operating license that will be retired.

Therefore Stakeholders contend that in determining whether a license renewal application can be approved all citing criteria promulgated in the NRC regulations for a new license must be included within scope, as a matter of law and fact.

Dr. P.T. Kuo, Director, Division of License Renewal Workshop clearly identified and acknowledged that the, ""License Renewal Rule"

allows a new licensed to be issued to operate for up to 20 years beyond the current 40-year term."

> License Renewal Workshop of Dr. P.T. Kuo, Director,
> Division of License Renewal, March 28-30, 2007
> Slide 4 Introduction
> - Atomic energy Act, as amended 1954
> -Allows for renewal
> -§10CFR54 - """"License Renewal Rule" allows a new licensed to be issued to operate for up to 20 years beyond the current 40-year term."

The NRC's relicensing rules state that the old license will be retired, and a new superceding license will be issued. The meaning of superseding, as defined by Webster's Revised Unabridged Dictionary "superseding as: Supersede \Su`per*sede"\ clearly means to replace, to put in the place of:

> 1. To come, or be placed, in the room of; **to replace.**
>
> 2. **To displace, or set aside, and put another in place of**; as, to supersede an officer.
>
> 3. **To make void**, inefficacious, or useless, by superior power, or by coming in the place of; to **set aside**; to render unnecessary; to suspend; to stay.

In the event the NRC approves Entergy's LRA, the NRC will retire the current licenses, and issue a new superceding license for a 20 year period. Therefore Stakeholders contend that all citing criteria promulgated

in the NRC regulations for a new license, as delineated in Regulatory Guide 4.7 - Appendix A - Site Safety Considerations for Assessing Site Suitability for Nuclear Power Stations, must be include within scope, as a matter of law and fact.

The criteria includes the following Regulations and Regulatory Guides which must be considered by the NRC prior to the issuance of a new license:

1. Geology/Seismology

Geologic and seismic characteristics of a site, such as surface faulting, ground motion, and foundation conditions (including liquefaction, subsidence, and landslide potential), may affect the safety of a nuclear power station. Including Relevant regulations 10 CFR 100.23 Geologic and Seismic Siting Factors", and Regulatory Guide 1.70, Chapter 2 (identifies safety-related site characteristics) Regulatory Guide 1.29 (discusses plant safety features which should be controlled by engineering design, Regulatory Guide 1.165 Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion, Regulatory Guide 1.132 Site Investigations for Foundations of Nuclear

Power Plant. (CONTENTION # 16 of this Petition is referenced and incorporated fully, as if set forth herein.)

Indian Point is located on the Ramapo fault. At the time of its initial siting 10 CFR 100.23 had not been finalized, and therefore a complete seismology evaluation was never done and/or completed.

2. Atmospheric Dispersion

The atmospheric conditions at a site should provide sufficient dispersion of radioactive materials released during a postulated accident to reduce the radiation exposures of individuals at the exclusion area and low population zone boundaries to the values in 10 CFR 50.34, including 10CFR Part 50, and Regulatory Guide 1.23 "Onsite Meteorological Programs", Regulatory Guide1.145 Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.

3. Exclusion Area and Low Population Zone

In the event of a postulated accident at a nuclear power station, radiological consequences for individual members of the public outside the station must be acceptably low, including 10 CFR Part 100, "Reactor Site Criteria," requires an "exclusion area" surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and

property, and a "low population zone" (LPZ) which immediately surrounds the exclusion area.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that at any point on the exclusion area boundary and on the outer boundary of the LPZ the exposure of an individual to a postulated release of fission products (as a consequence of an accident) be less than 25 rem total effective dose equivalent, for specified time periods. Regulatory Guides 1.3, 1.4, 1.5, and 1.25 give calculational methods,

4, Population Considerations

Locating reactors away from densely populated centers is part of the NRC's defense-in-depth philosophy and facilitates emergency planning and preparedness as well as reducing potential doses and property damage in the event of a severe accident. 10 CFR Part 100, "Reactor Site Criteria," requires the following:

An "exclusion area" surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property, and a "low population zone" (LPZ), which immediately surrounds the exclusion area.

The nearest distance to the boundary of a densely populated center containing more than about 25,000 residents must be at least one and one-third times the distance from the reactor to the outer boundary of the LPZ.

Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental,

economic, or other factors, which may result in the site being found acceptable.

(CONTENTION 17 of this Petition is referenced and incorporated fully, as if set forth herein.)

5. Emergency Planning

To ensure that adequate protective measures can be taken to protect members of the public in the event of an emergency, the characteristics of the site should not preclude development of such plans. 10 CFR Part 100, "Reactor Site Criteria," requires that:

> Site characteristics must be such that adequate plans to take protective actions for members of the public in the event of emergency can be developed.
>
> 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires:
>
> Reasonable assurance that adequate protection can and will be taken in the event of a radiological emergency.
>
> Emergency planning zones (EPZ) consisting of the plume exposure pathway EPZ with an area about 16 km (10 mi) in radius, and the ingestion pathway EPZ with an area about 80 km (50 mi) in radius.
>
> NUREG-0654/FEMA-REP-1, Rev.1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (November 1980),2 provides guidance on performing an ETE.

It is important to note that NRC does not have a clearly defined definition of "reasonable assurance" or of "adequate protection".

(CONTENTION # 18 and CONTENTION #20 of this Petition are referenced and incorporated fully, as if set forth herein.)

6. Security Plans

To prevent plant damage, and possible radiological consequences to the public as a result of acts of sabotage, the characteristics of the site should not preclude development of adequate security plans.

10 CFR 100.21(f) states that site characteristics must be such that adequate security plans and measures can be developed.

Also, 10 CFR Part 73, "Physical Protection of Plants and Materials," prescribes requirements for establishment and maintenance of a physical protection system for the protection of special nuclear materials at fixed sites and of plants in which special nuclear material is used.

(CONTENTION # 19 of this Petition is referenced and incorporated fully, as if set forth herein.)

7. Hydrology and .7.1 Flooding

Precipitation, wind, or seismically induced flooding (e.g., resulting from dam failure, from river blockage or diversion, or from distantly and locally generated sea waves) can affect the safety of a nuclear power station. 10 CFR 100.23, "Geologic and Seismic Siting Criteria"; Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"; Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Section 2.4); 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants;" Criterion 2, "Design Bases for Protection Against Natural Phenomena".

(CONTENTION # 49 of this Petition is referenced and incorporated fully, as if set forth herein.)

8. Water Quality

Contamination of ground water and surface water by radioactive materials discharged from nuclear stations could cause public health hazards. 10 CFR Part 20, "Standards for Protection Against Radiation"; 10 CFR Part 50, "Licensing of Production and Utilization Facilities".

The current ground water contamination at Indian Point must be fully evaluated and remediated to protect the public against radiation, prior to the issuance of a new license for 20 years. (CONTENTIONS #35, #36, & #37 of this Petition are referenced and incorporated fully, as if set forth herein.)

9. Industrial, Military, and Transportation Facilities

Accidents at present or projected nearby industrial, military, and transportation facilities may affect the safety of the nuclear power station. 10 CFR 100.21, "Non-seismic Siting Criteria"; 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4, "Environmental and Dynamic Effects Design Bases"; Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Section 2.21 (lists types of facilities and potential accidents); Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release".

It should be noted that the West Point Military Academy is less than 8 miles from Indian Point and Fort Smith is less than 3 miles, Additionally 8 major airports are located within 50 miles of Indian Point, and the FAA has

proposed adding 600 flights over Rockland County during the proposed new license period.

(CONTENTION # 50 of this Petition is referenced and incorporated fully, as if set forth herein.)

CONCLUSION

Public health and safety cannot and must not be grandfathered in for an additional 20 year period of licensed operation without properly evaluating the siting of the plant under NRC regulations.

Stakeholders contend that said NRC requirements for a new license place the siting criteria for a new plant, including but not limited to Population density, Emergency Plan and seismology, fully within scope in this LRA process.

Stakeholders further, contend that the NRC must apply all siting criteria, for a new license prior to the NRC's approval of the Applicant's LRA, and prior to the issuance of a superseding license for Indian Point 2 and Indian Point 3.

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CONTENTION 16: An Updated Seismic Analysis for Indian Point must be Conducted and Applicant must Demonstrate that Indian Point can avoid or mitigate a large earthquake. Indian Point Sits Nearly on Top of the Intersection of Two Major Earthquake belts,

Indian Point is situated virtually at the intersection of two major earthquake belts, each approximately 20 miles wide. One belt extends from Putnam and Westchester counties to New York City. The other extends southwesterly along the Hudson Highlands from Peekskill to near Reading Pennsylvania and is bounded on the southeast by border faults of the Newark Basin, including the Ramapo Fault. The two belts merge near Stony Point and Indian Point.

The region in which Indian Point is sited has, in fact, repeatedly experienced seismic activity. In 1737 there was an earthquake (magnitude 5.2). In 1783 there was an earthquake (magnitude 4.9). In 1884 there was another earthquake (magnitude 5.2). In 1980 there was a tremor centered in

Annsville (magnitude 2.9) near Indian Point, and in 1985 there was a tremor centered in Ardsley on the Dobbs Ferry fault.. In addition, a series of quakes that commenced in 2003 approximately 75 miles southwest of Indian Point are associated with the Ramapo fault system.

The fact that the region has not experienced a major earthquake in a long time does not mean that there will not be an earthquake which could cause extensive damage to Indian Point.

As stated by Dr. Leonardo Seeber, a seismologist at the Lamont-Doherty Earth Observatory of The Earth Institute at Columbia University (Lamont-Doherty): "The earthquake issue should not be discarded as insignificant. It is usually given less attention than it deserves because of low probability, but it has high impact." Notably, a few hundred years of history in the billion-year world of tectonic plate movement is too short a time span to project the future with much assurance. "Even where geology doesn't alert you to earthquakes," Dr. Seeber pointed out, "you can still have them." Interview with Dr. Seeber, reported in *The Journal News*, "Lessons learned from the earthquake in Japan," by Greg Clary (July 20, 2007.)

In a July 2004 presentation made at public hearings and submitted in writing to the NRC (Earthquake Risks to Indian Point Briefing), *Earthquake Risks to Spent Fuel at Indian Point, a Statement* by Lynn R. Sykes, Higgins

Professor of Earth and Environmental Sciences, Lamont-Doherty Earth Observatory of Columbia University at Meeting of U.S. Nuclear Regulatory Commission on Interim Storage of Spent Nuclear Fuel at Indian Point (July 15, 2004). Dr. Lynn R. Sykes, a seismologist at Lamont-Doherty and a professor of Earth and Environmental Sciences at Columbia University, noted that the east-northeasterly direction of maximum compressive stress in the Indian Point region is such that either faults of either orientation could rupture in significant earthquakes. Dr. Sykes noted that any estimate of earthquake risk necessarily includes both belts, which, by his calculus, results in a risk of 27 percent.

Neither Applicant Nor the NRC Has Conducted a Seismic Analysis Relevant to the Relicensing of Indian Point

Seismic hazard analysis is based on a combination of earthquake data, geologic data on active faults, and on tectonic modeling ("Seismic Analysis").

In his Earthquake Risks at Indian Point Briefing, Dr. Sykes observed that knowledge about earthquake risks at Indian Point has not been reviewed for over 30 years and is now outdated. Moreover, the design spectrum used

for Indian Point decades ago is very probably too low. In August 2004, the environmental watchdog group Riverkeeper wrote the NRC demanding that the agency take action to ensure that a Seismic Analysis for Indian Point was conducted. In June 2007, FUSE filed a complaint with the NRC demanding that a Seismic Analysis be conducted. In July 2007, New York's Attorney General Andrew Cuomo called on the NRC to include seismic and geographic issues in its consideration of Indian Point's relicensing.

Yet despite concerns expressed by experts, public interest groups and public officials, neither Entergy nor the NRC has conducted an updated Seismic Analysis of the Indian Point site, and the relicensing application ignores critical seismic issues. This is particularly irresponsible, as the operation of seismic monitoring stations near Indian Point have been discontinued and many of the originally installed seismic detection components installed back in the 1970s have ceased functioning and/or are no longer providing reliable data. Indeed, NRC inspector Mark Cox advised the North County News that there are no devices at Indian Point to monitor seismic activity. North County News "NRC to Indian Point: 'You're safe' but doesn't support ISA," by Abby Luby, May 4, 2007.

An Earthquake Could Cause Severe Damage to Indian Point and Cause a Radiation Release

Most significantly, Dr. Sykes said, the seismology research on the Northeastern American region indicates that a shallow earthquake is likely to produce the accelerations and shaking of a larger magnitude earthquake (Earthquake Risks to Indian Point Briefing). This is because an earthquake in the Northeast is likely to have seismic shaking of a higher frequency. Thus a smaller magnitude earthquake in New York could create damage equivalent to a larger magnitude quake in the western US (California) or Japan.

High frequency shaking presents a particular danger to a highly complex facility with numerous at-risk components and equipment systems. The danger is acute with respect to wirings, cables and critical piping and Indian Point systems which have been embrittled, corroded, or otherwise weakened by age.

An earthquake can readily compromise any number of components and systems resulting in the discharge of unacceptable levels of radiation into the environment. An earthquake may also damage backup safety

equipment, preventing safe shutdown, and/or damage a reactor's water-cooling system and trigger a meltdown.

Critically, an earthquake could potentially disrupt the integrity of the containment structure. At the present time, the condition of Indian Point's containment is unknown because Entergy sought, and was granted, a 5 year delay of the regularly scheduled inspection of the dome liner (i.e., from 2003 to 2008). In the dome line of Indian Point 2 rust has been identified, indicating that this critical failsafe structure may have deteriorated. Working with malfunctioning equipment could lead to errors like those made by the operators at Three Mile Island in March 1979.

Perhaps most significantly, unlike other kinds of accident scenarios (as opposed to attack scenarios), the widespread and concurrent damage an earthquake could cause, would prevent operators from interpreting the reasons for malfunctions with any degree of confidence.

Thus, under the chaotic conditions an earthquake would engender, not just the likelihood of system and equipment impairment, but the likelihood of making egregious error is incalculably heightened.

The Lessons of the Kashiwazaki Nuclear Plant, following the July 2007 Earthquake in Japan, Cannot be Ignored in the Indian Point Relicensing Process

The heavy damage sustained by the Kashiwazaki-Kariwa Nuclear Power Plant (Kashiwazaki plant) both during and following the earthquake that struck the Chuetsu region of the Niigata prefecture in Japan on July 16, 2007 dramatically underscores the vulnerability of nuclear power plants to seismic events.

Reported malfunctions, damages, and other problems included: (1) a transformer fire, which took hours to put out; (2) the dislocation of a water-tight seal in the reactor core cooling system; (3) loss of power at the control center of the plant's liquid waste disposal facility; (4) disrupted electrical connection at the plant's magnetic transformer facility; (5) malfunctioning of water intake screening pumps; (6) a duct knocked out of place in a major vent; (7) water leaks inside all reactor buildings; (8) blowout panels knocked down in turbine buildings; (9) an oil leak from low-activation transformer waste oil pipes; (10) an oil leak at the reactor water supply pump facility; (11) oil leaks from a damaged transformer and magnetic transformer facility; (12) oil and air leaks at switching stations; (13) cracks in the embankment of the water intake facility; (14) broken connections and a broken bolt at an

electric transformer; (15) land turned to mud under parts of the plant; (16) water leaks from the diesel generator facility; (17) a burst extinguisher pipe; (18) a burst condenser valve; (19) a burst filtration tank; (20) the toppling of hundreds of barrels of low-level nuclear waste, 50 of which had their lids knocked off resulting in the spilling of their radioactive contents onto the floor; (21) leaks of radioactive gases and liquids, including radioactive cobalt-60, chromium-51 and iodine (although the full extent of release into the environment is unknown because vital monitoring data was lost or not obtained); and (22) the plant's emergency hotline for firefighting efforts was unusable because the temblor damaged the building where the equipment was located and workers could not enter the building.

Aside from the sheer volume of problems, the Kashiwazaki plant illustrates the unexpected manner in which problems can emerge under earthquake conditions. For example, the earthquake shook a spent fuel pool, causing the pool to overflow. Radioactive spent fuel pool water then apparently flowed along electric cables protruding from a damaged floor, and then along an air conditioning duct, ultimately dumping radioactive water into the sea.

Undeniably, the problems experienced at the Kashiwazaki plant could have readily escalated into a nuclear accident disaster. The fact that

Kashiwazaki escaped calamity should not lull the NRC into believing that Indian Point would fare as well. Kashiwazaki should serve as a warning, and the NRC should take heed.

The Lessons of Yucca Mountain Cannot be Ignored in the Indian Point Relicensing Process

After 25 years of geological analysis, engineers working at the Yucca Mountain nuclear waste dump recently discovered that a fault line runs beneath planned storage pad structures for radioactive fuel canisters, some hundreds of feet away from where scientists had believed the fault lay.

This discovery illustrates the need to conduct a full analysis of the characteristics of the Indian Point site and presents further evidence that reliance upon outmoded seismology is foolhardy.

Special Risk Presented to the Spent Fuel at Indian Point Under Earthquake Conditions

A. Spent Fuel Pool Risk

As noted by the Lamont-Doherty seismologist Dr. Lynn R. Sykes, spent fuel assemblies are likely to be more sensitive to shaking at the higher frequencies likely in the Northeastern U.S. (Earthquake Risks to Indian Point Briefing). Thus the risk of an accident involving the high-density spent fuel pools would be elevated in the event of an earthquake.

Critically, earthquake risks to spent fuel pools were not even a point of consideration during Indian Point's original licensing hearings in the mid-1970s. Back then, the NRC operated under the assumption that the spent fuel pools were needed only for temporary waste disposition and the highly radioactive spent fuel would be shipped offsite. As with many other overly-optimistic NRC and industry assumptions, this one turned out to be false. Currently the Department of Energy concedes Yucca Mountain is unlikely to open for at least another decade.

Of particular concern is the fact that the density of nuclear fuel assemblies in the spent fuel pools has steadily increased over the past three decades, leading to ever greater risks from earthquake shaking. The National Academy of Sciences, the Sandia National Laboratory, and other preeminent scientists and nuclear experts have all warned of the catastrophic consequences that could result from spent fuel going critical and heating up

relatively rapidly to temperatures at which the zircaloy fuel cladding could catch fire, produce radioactive aerosols, and release massive quantities of radionuclides into the environment.

(Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report by National Academy of Sciences *(2006); Spent Fuel Heatup Following Loss of Water During Storage* by Allan S. Benjamin et al, Sandia National Laboratory (NUREG/CR-0649, SAND77-1371, 1971). See also, *Reducing the Hazards from Stored Spent Power-Reactor Fuel in the United States* by a multi-institutional team of researchers led by Dr. Frank von Hippel, co-director of the Program on Science and Global Security at Princeton University (2003); *Robust Storage of Spent Nuclear Fuel: A Neglected Issue of Homeland Security* by Dr. Gordon Thompson, director of the Institute for Resource and Security Studies (January 2003){Exhibit MM}).

Patently, in an earthquake, the damage to Indian Point's spent fuel pool could result in an exothermic reaction sufficient to start an uncontrollable nuclear fire. This consequence could occur because of spent fuel pool structural cracking, displacement of water from falling debris, rapid water vaporization due to chemical reactions or fire and/or the destruction or blockage of cooling intake structures.

B. Independent Spent Fuel Storage Installation Facility Risks

As Indian Point spent fuel pools are rapidly filling up, Applicant plans to store the overflow of high level nuclear waste in 78 Holtec International HI-STORM 100S(B) Casks (Holtec Casks), to be arranged in a 6 X 13 array on a 19,968 square foot concrete pad. Each Holtec Cask is approximately 20 feet tall. Entergy has indicated it intends to transfer 3,000,000 pounds of high level nuclear waste to this Independent Spent Fuel Storage Installation Facility (ISFSI).

The ISFSI presents two risks which are ignored by Applicant.

The first is that the Holtec Casks are designed to withstand only a 4.5 quake, which is less than the design basis of 5.5. As noted by Dr. Sykes, dry casks, like the spent fuel assemblies, are probably be more sensitive to shaking at higher frequencies (Earthquake Risks to Indian Point Briefing). Thus even the design basis of 5.5 is likely inadequate.

The second is that, under the conditions of an earthquake, the planned design of the ISFSI allows a bowling pin dynamic to occur whereby one Holtec Cask tipping over can topple another, and so on and so on, until a

large number are toppled. (This may have been the scenario at the Kashiwazaki plant, where hundreds of barrels of low-level nuclear waste fell over, 50 of which spilled their radioactive contents.) Incomprehensively, Applicant does not even plan to take the basic precaution of having the Holtec Casks bolted to the ISFSI's concrete pad foundation.

C. The Spent Fuel Pool and ISFSI Risks are Additive, and Most Likely Multiplicative, Under Earthquake Conditions

In a July 12, 2004 statement of concern by the Union of Concerned Scientists (Statement of Concern), David Lochbaum, a nuclear safety engineer, warned of a fundamental flaw with the Indian Point plan for spent fuel storage. The Statement of Concern notes that the ISFSI adds to the overall risk profile for the Indian Point site, since it is not intended to alleviate the risks inherent in spent fuel pool storage, but to enable greater quantities of spent fuel to be stored on site. Specifically, the Statement of Concern states:

> The consequences of a spent fuel pool accident depend on the inventory of radioactivity in the pool. Simply put, the more there is, the more that can be released. Thus, the "older" spent fuel in

> the pool may have little impact on the probability of an accident but it has considerable impact on the consequences from an accident.
>
> The risk from a spent fuel pool loaded to near-capacity approaches its maximum value. The freshly discharged spent fuel defines the accident's probability while the collective sum of spent fuel in the pool defines the accident's consequences. That risk is maximized when the pool is maintained near full capacity.
>
> The ISFSIs add to the spent fuel pool risk by introducing another spent fuel storage accident scenario: namely, the damage to spent fuel assemblies located in the dry casks...The probability of a dry cask accident increases with the number of casks loaded and placed in the ISFSI.

Thus, the ISFSI adds to the spent fuel pool risk. Manifestly, the risk of simultaneous damage to spent fuel pool and ISFSI structures is a real possibility in an earthquake scenario. (This was, in fact, the case at the Kashiwazaki plant.) This means that such risk must be viewed as multiplicative, since earthquake conditions most certainly would greatly stress plant manpower resources, create a substantial level of disorientation (since many malfunctions will be occurring at once), and reduce the ability of off-duty personnel and first responders to rapidly get to the site. Under such conditions it would be false to assume adequate mitigative measures could be taken.

Moreover, in light of the inability of the federal government to establish a long term repository for nuclear waste at Yucca Mountain, and

given the fact that, even if it opens, Yucca Mountain will not be able to take any of the nuclear waste generated by Indian Point after 2011, it must be postulated that thousands of tons of radioactive waste will be stored on site for foreseeable future, and possibly for generations. The duration of this risk must be incorporated into the overall risk calculus.

Indian Point is an Aged Plant, Especially Vulnerable to the Risks Presented by an Earthquake, and Applicant's Aging Management Plan Ignores these Risks

Stakeholders contend that the effects of or associated with aging, including embrittlement, corrosion, rust, heat, exposures to chemical agents and constant radiological bombardment (Aging Effects) have destabilized and weakened the tensile strength of the reactor and critical equipment components and systems to a point where there may exist an unacceptable risk of break up under the violent conditions of a seismic event.

The numerous problems Indian Point has experienced over the past several years, most notably the spent fuel and other leaks, which are strongly indicative of age-related degradation.

Indian Point's aged piping is of particular concern. (Indeed, in August 2004 an earthquake in Illinois broke an underground pipe attached to one of the Dresden nuclear power plan's condensate storage tanks, resulting in a leak of tritium into the groundwater.) Since there is literally miles of aged piping at Indian Point, and much of it is buried underground, encased or otherwise difficult to access and inspect, it is prudent to assume that pipe ruptures during a major seismic event could either directly or indirectly consequence a major accident.

It is beyond cavil that existing studies on Aging Effects are woefully incomplete. But studies on high pressure/high heat steam boiler systems failures and explosions provide evidence that a full analysis of how Indian Point's aging components and systems would function under high-frequency vibration conditions is warranted.

The risk of an earthquake is real, but is callously disregarded in Applicant's aging management plan.

The NRC Must Apply New and Significant Information Regarding Earthquake Risk, Rather than Relying on Outdated and Inadequate Criteria

The NRC has claimed that nuclear power plants are designed under very stringent requirements to assure they can safely shut down following design basis events such as large fires and earthquakes. This claim is meaningless unless substantive evidence demonstrates that Indian Point can avoid a radiation release accident in the event of a large magnitude earthquake.

Recent events (i.e., the Kashiwazaki plant and Yucca Mountain) have revealed the risks inherent in reliance upon insufficient, decades old quake-resistant design standards and outdated geological research.

Substantial evidence exists to demonstrate that present design basis criteria are outmoded and the estimated 100 year duration for an earthquake of great magnitude is inapposite. The NRC cannot simply discount modern seismology.

Before Indian Point is relicensed, the NRC must require that the margin of earthquake-resistance safety at Indian Point be in accord with the most recent scientific knowledge, and not simply in compliance with regulatory criteria that is patently outdated and inadequate.

Conclusion

The seismic design of Indian Point, including its spent fuel pool and planned dry cask nuclear waste facilities are based on standards that are too old from the viewpoint of modern seismology.

An updated Seismic Analysis for Indian Point must be conducted by a recognized and (most importantly) independent authority on seismic research. In order to obtain a new superseding license, Entergy must be able to demonstrate that Indian Point meets both the seismic criteria set forth in 10 CFR Part 100.23 and the standards of modern science.

The prime mandate of the NRC under the Atomic Energy Act is to protect public health and safety. All NRC regulations and guidances are subservient to this prime mandate and cannot be utilized as a means to thwart it.

Thus the seismic design basis of Indian Point may not legitimately be grandfathered in for the sake of allowing the plant's continued operation.

CONTENTION 17: The population density within the 50 mile Ingestion Pathway EPZ of Indian Point is over 21 million, the population within in the 10 mile plume exposure pathway EPZ exceeds 500,000.

Indian Point is surrounded by one of the most densely populated area in the United States, 21 million people live within 50 miles of Indian Point. Based on 10 CFR Part 100" Reactor Site Criteria" Indian Point could not be

cited where is it located today in Westchester County. Therefore the NRC cannot issue a new license for Indian Point.

In 2006 the immediately surrounding area had substantially more than 25, 000 residents, in fact the communities directly adjacent to Indian Point had 84,848 residents: Peekskill 24,601, Buchanan 2, 269,. Croton-on Hudson 7,899, Stony Point 14,975 and Haverstraw 35,104. Reactor sites should be located away from very densely populated centers. Areas of low population density are generally preferred. The projected Population increase during the new superseding 20 year license period distinguishes Indian Point from any other plant in the nation. The population surrounding Indian Point has exponentially increased from 1960 to 2006, 49%. The population in the surrounding Counties are continuing to grow rapidly. In fact, Orange County, is the fastest growing county in New York State. Using the same rate of increase, the estimated projected population in the counties surrounding Indian Point will be between 50%- 63% increase by 2035. (See Census Study Exhibit NN)

Public health and safety cannot be grandfathered, especially in light of such substantial changes in population density. Population density increases directly affect the ability to evacuate the communities surrounding Indian Point. At the time the plants were first site, the area was primarily

farmland. The operators of the plant has the ability to mitigate the rapidly increasing population surrounding the plant, by purchasing the necessary acreage in order to maintain the require LPZ. They did not, so today the population density surrounding Indian Point is the largest in the nation. The U.S. census population projections to 2030, suggests that during the new 20 year license period the population surrounding Indian Point will keep expanding.

CONTENTION #18: Emergency Plans and evacuation plans for the four counties, surrounding are inadequate to protect public health and safety, due to limited road infrastructure, increased traffic and poor communications.

On March 7 , 2003, former FEMA director James Lee Witt, who was hired by the State of New York to evaluate the Emergency Planning for Indian Point, have determined that the current evacuation plan is inadequate, unworkable and unfixable, due to the limited road infrastructure and enormous population density surrounding Indian Point.

A. Stakeholders contend that the Emergency plans do not fulfill their purpose, to provide reasonable assurance of public health and safety . James Lee Witt, former FEMA director, who was called into the establish order after Katrina, evaluated the emergency plans for New York State and found

them to unfixable. (Witt Report Executive Summary Exhibit OO) Local and New York State authorities have refused to certify these unworkable emergency plans for the past five years. The adequacy of the Emergency Plans must be considered in the LRA before a new license can be approved. Additionally the Emergency plan must also included in the Environmental Impact Statement of the Applicant's LRA..

The NRC acknowledges this in the following statement:

> "For operating power reactors, 10 CFR 50.54(s)(2)(ii) requires that "If ... the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency ... the Commission will determine whether the reactor shall be shut down until such deficiencies are remedied or whether other enforcement action is appropriate."

Adequate Emergency Plan is a requirement and an important part of the issuance of a new (superceding) nuclear plant operating license. as per § 50.47, "Emergency Plans," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," paragraph (a)(1) which states no operating license for a nuclear power reactor :

> "will be issued unless a finding is made by the NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency."

Stakeholders assert that the proposed new license cannot be issued because the Emergency Plans for Indian Point are inadequate to protect public health and safety.

The traffic study prepared by KLD Associates, the traffic consultants hired by Entergy, show it will take nearly twice as long to evacuate people from the 10 mile region around Indian Point as previously believed, due the increase in traffic and population, with little improvement in infrastructure. It could take up to four hours to mobilize residents and up to 10 hours to evacuate the region in good weather. Traffic congestion from "shadow evacuations" would increase those times, the report found. As the population continues to grow over the next 20 years, traffic congestion will only get worse, and evacuation times will continue to increase.

On the eve of Katrina, the Nuclear Energy Institute, (NEI) the powerful lobbying group for the nuclear industry, submitted a white paper to the NRC recommending reduction of the evacuation area of 10 miles to a 2 mile wedge, due to associated costs to the industry and feasibility of evacuation due to the increase population. In the NEI's recommendations the majority of the surrounding population would be sheltered-in-place, thereby exposing individuals to potentially more than 25 rem, does not adequately protect the public from exposure to radiation. Especially given

the reality that the vast majority of home and buildings are wood frame and only provide less than 40% protection from airborne radioactive contamination.

FEMA recognizes this concern in their February 21, 2003 report on emergency preparedness at Indian Point. On page 6 of Attachment B of the report, FEMA states:

> NUREG -0654, Appendix 1 issued in 1983 and enhanced in 1996, in the NRC Supplement 3 to NUREG-0654.FEMA-REP1 "Criteria for Protective Action Recommendations for Severe Accidents. States that "Since the publication of the original guidance extensive studies of severe reactor accidents have been performed. These studies clearly indicate that for all but a very limited set of conditions, prompt evacuation of the area near the plant is much more effective in reducing the risk of early health effects than sheltering the population in the event of severe accidents. In addition, studies have shown that except for very limited conditions. Evacuation in a plume is still more effective in reducing health risks that prolonged sheltering near the plant. The NRC and FEMA recommend that the population near the plant should be evacuated."

Over an additional 20 years the population density in the area surrounding Indian Point is projected to continue to increase., Therefore based on Entergy's own recent traffic study and the inadequacy of sheltering in place there can be no "reasonable assurance that public health and safety will be protected during an additional 20 years of operation, which is in direct contradiction with the NRC and FEMA regulations. Schools, reception centers, and hospitals are not equipped with food, medicine, water,

decontamination equipment, and basic supplies necessary for even short term sheltering.

This is no longer an honest emergency evacuation plan. Entergy has not taken any significant actions to improve emergency plan since James Lee Witt found the plans to be insufficient, nor does Entergy delineated aging management of the emergency plan over the next 20 years in the LRA, as the surrounding population continues to increase.

Sheltering-in-place is not an acceptable alternative to evacuation and does not constitute a function emergency plan. The NRC is charged with a mandate to protect public health and safety, but instead by rubber stamping the NEI';s white paper to shelter-in-place, they are instead protecting the profits of the industry, over the lives of the people.

Therefore until there is "reasonable assurance that adequate protection can and will be taken in the event of a radiological emergency, the NRC under its own guidelines, cannot issue a superceding license for an additional 20 years.

B. In the alternative, Stakeholders asserts that a comprehensive evaluation of any and all resulting Environmental Impacts and Costs of such accident pathway caused by failure of the Emergency Plans must be

included in the EIS of the LRA and are not requesting that the entire adequacy of the evacuation plan be placed within scope of the EIS.

Stakeholders assert that any and all costs of a failed Emergency Plan do rightfully belong in the LRA, including a full complete record of those potential costs as set forth in various scientific studies including, but not limited to, the Witt Report -Exhibit OO, and Westchester's County COWPUSA report.

The NRC acknowledged that shutdown risk associated with shutdown and refueling (however remote) can occur:

In January 1992, the Nuclear Utilities Management and Resource Council (NUMARC) issued Revision 2 of NUMARC/NESP-007, "Methodology for Development for Emergency Action Levels (EALs)," 2 which contained guidance on Emergency Action Levels (EALs) development that accounted for lessons learned from ten years of using the NUREG-0654 guidance. The NRC stated in Revision 3 of Regulatory Guide 1.101 (August 1992), that Revision 2 of NUMARC/NESP-007 was considered to be an acceptable alternative to the guidance provided in NUREG-0654 for development of Emergency Action Levels (EALs) to comply with 10 CFR 50.47 and Appendix E to 10 CFR Part 50..

In addition, the NRC stated in Revision 3 of Regulatory

Guide 1.101 that there is a likelihood that the results of ongoing risk studies related to shutdown may necessitate revision of both the NRC Emergency Action Levels (EAL) guidance (NUREG-0654) and the NUMARC EAL guidance (NUMARC/NESP-007). Appendix E to 10 CFR Part 50 specifies that:

> Emergency Action Levels are to be used as criteria for determining the need for taking emergency response actions (e.g., notification of emergency response organizations). The need for emergency response actions depends on the degree of degradation of plant safety during an event. The shutdown risk studies have demonstrated that events warranting emergency classification and response (although very unlikely) can occur in the shutdown and refueling mode of plant operation.

The above passages are from, Regulatory Guide 1.101 Emergency Planning and Preparedness for Nuclear Power Reactors". The NRC admits there are events that can occur that would require implementation of the Emergency Plan.

In the event the Emergency Plan is implemented, there is also the possibility of failure of the plan to perform adequately in the intended activation scenario.

Therefore, the potential Environmental Costs and Impacts of such failure must be transparently evaluated and considered in the EIS of the Applicant's LRA to ensure that adequate protective measures can be taken to

protect members of the public in the event of an emergency. The characteristics of the site should not preclude development of such plans.

10 CFR Part 100, "Reactor Site Criteria," requires that:

> Site characteristics must be such that adequate plans to take protective actions for members of the public in the event of emergency can be developed.

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires:

> Reasonable assurance that adequate protection can and will be taken in the event of a radiological emergency. Emergency planning zones (EPZ) consisting of the plume exposure pathway EPZ with an area about 16 km (10 mi) in radius, and the ingestion pathway EPZ with an area about 80 km (50 mi) in radius.

NUREG-0654/FEMA-REP-1, Rev.1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (November 1980), 2 provides guidance on performing an ETE. It is important to note that NRC does not have a clearly defined definition of •reasonable assurance or of •adequate protection.

Sheltering in place is not an acceptable substitute, however the NRC, based on NEI's white paper, has reduced the standards despite,

CONTENTION: 19 Security Plans

Stakeholders contend that the way the force-on-force (FOF) tests are conducted do not prove that the Indian Point security force is capable to defend the facility against a credible terrorist attack or sabotage. The LRA does not address how Security, as required under section 10 CFR 100.12(f) and 10 CFR Part 73, will be managed during the proposed

additional 20 years of operation against sabotage/terrorist forces with increasing access to sophisticated and advance weapons.

In a letter dated 12/9/03 by Project on Government Oversight to the NRC then Chairman Diaz, the following issues that must be addressed to insure adequate security were raised (Exhibit PP):

Dumbed-Down Design Basis Threat (DBT) – It has been widely reported in the press that prior to 9/11, nuclear power plants were required to have defenses designed to protect against only a ridiculously small attacking force – three terrorists. In contrast, the intelligence community generally believes that terrorists would attack a target with a squad-sized force, which in the Army special forces is 12 and the Navy Seals is 14. In other words, the NRC would need to at least quadruple its old DBT.

NRC's new DBT, does not reach the 12 to 14 level that is appropriate. Representatives of other federal agencies have told Project on Government Oversight that the NRC's new DBT remains inadequate. (POGO Letter Exhibit PP)

The NRC argues that the new DBT is the largest threat against which a private security force can be expected to defend. This rationale is

backwards and conflates two separate considerations – what is the size of the threat and what should the nuclear power industry be required to do to in the face of such threats.

The NRC policy decision to limit the size of the DBT (under terrific pressure from the NEI and its friends in Congress) was based mainly on its assessment of what is reasonable to ask of a private force.

That approach ignores the most fundamental question: what is the credible threat against the facilities? The size of the DBT must be based on that threat. Furthermore, NRC's justification of its too-low DBT rings hollow, as the Department of Energy (DOE) also relies on a private security force, yet at some facilities, DOE claims to protect its facilities against twice as many terrorists as the NRC does.

Under Use of Readily-Available Lethal Weapons – It is well known in security circles that there are weapons that are available to terrorists that can penetrate bullet-resistant enclosures (BREs), which are quasi-guard towers. BREs are included in the defensive strategy of a number of nuclear power plants, including Indian Point. Some time ago, the Department of Energy abandoned the use of its state-of-the-art guard towers (which are far more robust than most BREs) because of their vulnerability to readily-available weapons. Indian Point officers have been aware of the controversies

surrounding BREs and have brought their concerns not only to Entergy, but also to the NRC Region I, with no response at all.

Several years ago, the DOE developed a classified official Adversary Capabilities List which includes weapons and explosives that are readily available to terrorist groups. The NRC should review this list and ensure its Design Basis Threat includes them. For example, .50 caliber sniper rifles (which have been available since World War I) and Armor-Piercing Incendiary rounds (which are available in gun shops for $1 per round) made the DOE guard towers so vulnerable they were abandoned. Other weapons were also of concern, including the rocket-propelled grenades which have been used frequently by near-children around the world in war-torn countries, with great success against hardened targets, and which could easily be launched from boats on the river, or from unpatrolled shores.

Unrealistic Timing and Location of Attack – It appears the NRC conducted the three FOF tests at Indian Point during the daylight at the beginning of the night shift, and began at least two of the tests in the owner-controlled area. There are several problems with this:

* The security force being tested had just come on duty and was not yet fatigued by a 12-hour shift, hours typically worked by Indian Point security officers five to six days a week.

* The security officers knew within the hour that the test was to begin, as the day shift was held over an extra hour to cover as a shadow force so that the night shift could be tested at the beginning of their shift.

* It is widely believed in the intelligence community that no one will attack during daylight, as it is to the attacker's advantage to have the cover of darkness. Despite this, all three FOF tests occurred between 4-6 pm. Furthermore, in two of the three tests, the mock terrorists were required to cross open fields in broad daylight in order to reach the protected area, making it that much easier for them to be observed by the security officers.

* The mock terrorists attacked from only one entry point. In addition, the NRC and Entergy agreed that, if the attackers were successful in reaching the protected area fences, there would be a halt in the action and the adversaries would be brought inside of the fences (to prevent any actual damage to the fences during the exercise) – making it perfectly obvious from

where the attack will be coming. POGO had previously alerted the NRC to a particular vulnerability involving the fences at most nuclear facilities and was assured that this vulnerability would be taken into account in future FOF tests. However, it was does not appear to have been taken into account during the Indian Point force-on-force (FOF).

Amateur Mock Terrorists – A terrorist group has advantages that cannot be replicated in even the best mock attack FOF. However, the following limitations could have been partially ameliorated by the NRC, but were not:

* No Surprise. The security force knew for months in advance that this test was going to occur, training specifically for the approved scenarios. They even knew within minutes that the test was to occur, because of all the visiting dignitaries and the fact that they had strapped on Multiple Integrated Laser Engagement System (MILES) equipment..

* No Violence of Action. During a mock FOF there is no real danger – no live ammo, no colleagues dying or being maimed or any other adverse impact that would normally create chaos and in some cases cause the

protective forces to panic. As a result, security forces develop "MILES bravery."

* Safety First. The FOF tests are not conducted at high speed because of the overriding safety concerns. Therefore, people and vehicles are not going full tilt the way they would during a real terrorist attack, giving the protective forces time to pause to make decisions – time that they wouldn't have in a real life situation. Safety was also used as the reason for not conducting the tests at night. Sources told us that Entergy was worried participants could trip over rocks or step on snakes.

* No Trained Adversaries. The mock terrorists were security officers from another nuclear plant who had no training as adversaries. This training is critically important because it teaches the mock terrorist how to think and act offensively, as a real terrorist would, rather than defensively as a security guard would. Here again, both DOE and the military use trained adversaries to test their security forces.

The Security Forces Are On Their Own – It should be recognized that although the exercise was observed by the State Police and FBI, these law

enforcement entities cannot respond to an attack with SWAT capability before it is too late. Insofar as we know, these response times have not been tested at Indian Point. But tests at other facilities have shown that an attack is generally won or lost in between three and eight minutes, while it generally takes an hour or two for SWAT teams to respond.

Poor Planning: Lives at Risk – One of the FOF tests was quickly aborted when Coast Guard personnel, who had not been previously informed that the test was to occur, threatened to use their live ammo against the mock attackers. It is unacceptably poor planning to allow this kind of lack of professionalism, putting lives at risk.

Very few of the above issues have been addressed since POGO first put the NRC on notice in 2003, (Exhibit EE) and the LRA does not address any aging management plans with regard to enhancing the security force during the proposed 20 year superceding license.

The world is not becoming a safer place, but a much more dangerous one. Stakeholder contend that the NRC cannot approve a superceding 20 year license for Indian Point 2 or Indian Point 3 without first requiring the

Applicant to substantially improve security and develop an aging management plan to maintain security.

CONTENTION # 20: The LRA does not satisfy the NRC's underlying mandate of Reasonable Assurance of Adequate Protection of Public Health and Safety.

Statement of Issue:

Stakeholders contend that the Applicant LRA does not satisfy NRC's underlying mandate of reasonable assurance of adequate protection of public health and safety during the 20 year new license period.

On June 12, 2006, Richard S, Barkley, of the NRC wrote that "the NRC's definition of **Reasonable Assurance of Adequate Protection of Public Health and Safety** was stated in the Director's Decision, in the matter of Docket No. 50-346 (License No. NPF3) FirstEntergy Nuclear Operation Company (Davis-Besse Nuclear Power Station, Unit 1, April 22, 2004, 59 NRC 215), and NRC case law, to be, as a general matter, defined by the Commission's health and safety regulations themselves. There is reasonable assurance of adequate protection of public health and safety when the applicant or licensee demonstrates compliance with the

Commission's regulations. The regulations were established using defense-in-depth principles and conservation practice." (Exhibit MMM)

When failure to meet a commitment results in a violation of the Commission's health and safety regulations, the Staff is required to take the appropriate enforcement actions.

Unfortunately the NRC has not taken appropriate enforcement actions to protect the public health and safety, now or in the past. In a fact at Indian Point the NRC has granted seemingly endless exemptions, exceptions and deviations from its regulations, without enforcement consequences or only minor fines assessed on the operator. A recent example is Entergy's failure to install functional, required back-up powered siren system, that is required under the Energy Policy Act of 2005 to protect public health and safety.

On October 4, 2007, after the LRA has been accepted and Stakeholders were preparing Intervener Contentions, the NRC granted an exemption from fire safety standards, from 1 hour to 24 minutes, reducing the ;p Public notice of this significant change in the risk to public health and safety without public notice. (CONTENTIONS # 4 ,#5, #6, #7, #8, #9, #10 of this Petition are referenced and incorporated fully, as if set forth herein.)

The NRC issued a Confirmatory Order in January 2006 requiring the installation of back-up power for the siren system at Indian Point by Jan. 30, 2007. In January 2007, Entergy requested and received an extension but missed that deadline of April 15, 2007. The NRC merely fined Entergy $130,000 and extended the deadline to August 24, 2007, this new deadline has also been missed. To date, the functional siren system does not exist, and the Stakeholder, along with 20 million people in the 50 miles radius live every day at risk, due to the NRC and Entergy failure to follow the law.

Another example is the ongoing leaks, that were first found by accident by an independent contract, not through routine inspections. The source of the leaks has still not been identified, nor the total volume, but it is known that an enormous quantity of radioactive effluent has released unmonitored into the groundwater, into New York State's discharge channel and into the Hudson River, contaminating the property of the citizen's of New York and once again directly affects public health and safety.

In addition since it is a fact that Indian Point 2 and 3 were built to industry guidance, instead of NRC regulations, the standard of Reasonable Assurance of Adequate Protection of Public Health and Safety is

meaningless at Indian Point. Therefore the entire population, of 20 million residents, 8% of the United State population, living within the 10 mile emergency evacuation zone, 17.5 mile peak injury zone, and the 50 miles ingestion zone are all affected by the NRC's inability to maintain an enforceable standard of Reasonable Assurance of Adequate Protection of Health and Safety.

The magnitude of the impact on the effected population is LARGE, as the impact of the NRC not enforcing its own required standard regulations has a significant adverse affect on the population. This is evidenced by the fact that the State and County government surrounding the plant have found the emergency evacuation plans to be wholly inadequate. Robert Stephan, Homeland Security's Assistant Secretary for Infrastructure Protection reported in the Journal News, March 23, 2006 that, "The Nuclear Regulatory Commission has ranked Indian Point in terms of potential human consequences as the No. 1 site in the nation."' The issues surrounding Indian Point are unique, and the GEIS does not adequately address the site specific and unique issues of Indian Point to give Reasonable Assurance of Adequate Protecton of Public Health and Safety including but not limited to the following:

A. The population mass within a 50 mile radius of Indian Point exceeds 20

million citizens, 6% of the U.S. population, and is located in the most densely populated area surrounding a nuclear facility in the nation.

B. New York City, located 25 miles from the plant, is the hub of America's Financial institutions. A significant nuclear incident (accident) or terrorist attack on the facility that led to off site migration of radiological contaminants would be catastrophic in nature to not only the surrounding region, but to the entire nation, as it could quickly lead to Environmental Costs in excess of half a trillion dollars and could bankrupt America.

C. West Point Military Academy, the training ground for America's future leaders, and a vital American brain trust, which includes a U.S. mint, it located less than 8 miles away.

D. Indian Point is the only reactor site that is leaking radioactive strontium 90 into the ground, groundwater and Hudson River.

E. Indian Point is located on an active fault line, the Ramapo fault. F. On 9/11 at least one of the hijacked planes flew directly over Indian Point 2 and 3 before it destroyed the World Trade Center. G. Since 9/11 Indian Point is considered one of the most attractive and vulnerable terrorist targets in the nation.

H. In addition, the Indian Point site already has numerous non-compliance issues that place it in violation of NRC Rules and Regulations, with said

issues that are already contaminating the environment, and increasing the risk to the general public. These risks include, but are not limited to:

i. Numerous members of Congress, and a majority of the elected officials and local communities, question whether Indian Point is safe, and have repeatedly called for, and asked the NRC for an Independent Safety Assessment (ISA) (Exhibit AA).

ii. Despite various extensions granted by the NRC, Entergy has yet to come into compliance with NRC regulations as relates to having a working siren system. FEMA recently failed the system, and a full review of Entergy's own documents shows that the system ordered and installed FAILS to meet the Design Basis Criteria. Further, the old system, as NRC records show, also fails to come close to being in compliance with 10 CFR Rules and Regulations.

iii. The State and County governments within the 10 mile Emergency Evacuation Zone have stated it is there own belief that the Evacuation Plan is fundamentally flawed, and the Witt Report supports their conclusions. It is pointed out here, that the Emergency Plans tells us, "When you hear the sirens... go inside and follow instructions. However FEMA has admitted the Siren level is inadequate and therefore the sirens cannot be heard.

iv. Significant spent fuel pool leaks at IP1, IP2 and IP3, are leaking

strontium 90, cesium 137 and tritium. All the spent fuel pools at Indian show clear evidence of serious aged related degradation. Yet, since 2005 Entergy has been unable to locate, identify, stop and remediate said leaks.

v. A recently discovered leak at IP2, that was incorrectly categorized as a conduit leak, was in fact a leak in the fuel transfer tube.

vi. Entergy has been unable to locate and identify the leaks associated with reactor cooling systems which were only accidentally discovered when workers saw steam rising through the black top.

I, There are known Tritium, Strontium 90 and Cesium 137 plumes under the entire reactor site that are rapidly migrating towards the Hudson River. Said leaks represent a minimum of 250,000 gallons of radiological contaminants that are polluting the potable water resources of New York State, in violation of New York State Law. Such leaks have been, and continue to be, unmonitored in violation of the NRC own regulations and New York State law.

J. Both reactors are suffering severe BAC (Boric Acid Corrosion) of the reactor vessel heads...in fact, the corrosion issues are significant enough that Entergy has a standing order for new reactor vessel heads for IP2 and IP3 with delivery slated for 2011 and 2012 respectively. In order to install these vessel heads, it is probable that containment will have to be breeched.

K. IP2 is one of the few reactors in America to have suffered a significant Tube Rupture. This occurred back in 2000. Further, a recent Industry study has shown that tube fouling becomes a significant safety issue in pipes adjoining plugged pipes. Indian Point 2 and Indian Point 3 together have literally hundreds of plugged pipes in the reactor cooling system.

L. The series 400 stainless steel roller bearings on the traveling water screens for IP3 have huge holes, which is believed to be caused by corrosive microbes or lack of maintenance, This condition has existed since 1991, yet remains un-remediated.

M. One of the steel containment plates at Indian Point is failing. .

Indian Point cannot meet the Fire regulations of 10 CFR, and, in fact. Entergy has just requested the NRC further lower the Safety Margins although they were already granted exemption from the rules and regulations

N. Due to the closure of Barnwell, the low-level radioactive waste site, Entergy is planning to turn Indian Point into a low level radioactive site, without proper application and review.

O. Due to the failure of approval of Yucca Mountain, the spent fuel produced by Indian Point, which by regulation is only to be stored on site on an interim, temporary basis, will now become indefinite and potentially

permanent.

P. The Decommissioning Trust Funds for IP1, IP2 and IP3, are insufficient to restore the site, especially in light of the multiple leaks first noticed in 2005.

Mitigation measures with regard to Reasonable of Public Health and Safety would be warranted for impacts that would have the same significance level for all plants. However, due to the unique facts and issues at Indian Point, such mitigation must be site specific. Stakeholders content the LRA does not offer an aging management plan that will give Reasonable of Public Health and Safety at Indian Point, and therefore the NRC must deny the Applicant's LRA.

CONTENTIONS 22 -25 Indian Point was not required to comply with federally approved General Design Criteria, which constitutes a clear and flagrant violation of the Administrative Procedures Act, and Entergy's LRA fails to remediate the error, leaving Indian Point without adequate safety margins and the New York Metropolitan region without adequate assurance of protection of public health and safety

Issue Statement: Applicant has violated the regulatory rules for obtaining a new superseding license, as delineated in the Code of Federal

Regulations (CFR). Specifically, Applicant has failed to comply with the rules under 10 CFR 54, "License Renewal," including aging management as delineated under 10 CFR 54.21, but has instead based its License Renewal Application (LRA) on criteria promulgated by the nuclear trade industry.

Entergy's predecessors in interest in the operation of Indian Point's three reactors, Con Edison and the New York Power Authority (individually and collectively referred to as, Licensee) misrepresented the specific General Design Criteria (GDC) which formed the basis of the Safety Evaluation Report granting the licenses (hereinafter referred to, for simplicity, in the singular as license) for Indian Point's operation and subsequently remained in violation of the terms of the operating license and with federal rules for decades. Entergy never corrected the obvious error—placing economics ahead of the health and safety of the public.

Applicant, with the acquiescence of the NRC has flagrantly violated the Administrative Procedures Act, and as a result now has filed a fatally flawed LRA.

The Aging Management Programs proposed by Applicant are based upon misrepresentations of the actual General Design Criteria to which Indian Point was licensed. The as-built construction of the facility does not comply with the safety evaluation report, the operating license, or the CFR.

The 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 Atomic Energy Commission (AEC or Commission) originally evaluated a majority of the SEP-III plants before they issued Regulatory Guide (RG) 1.46 in May 1973 (AEC, 1973b). Although the Commission 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 by the AEC and the NRC over almost four decades for Indian Point.

Beginning in 1968, the AEC acted in direct defiance of the Administrative Procedures Act by approving Amendment Nine of the Operating License (contained in exhibit 1), in which Licensee acknowledged commitments to *trade comments* to draft General Design Criteria for its new plant. In addition, Licensee committed to trade comments to the proposed GDC 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).

Licensee claimed adherence to a General Design Criteria required for the licensing of Indian Point, and committed to such GDC in the 1970 Safety Evaluation Report (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 AEC under the 1970 Safety Evaluation Report (See Exhibit K), completely bypassing the federal 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 continued, when the Commission quietly authorized by retroactive fiat that the licensing process proscribed under federal rules for Indian Point could remain in violation of law. This series of events is evidenced by close examination of documents cited or submitted in the LRA. The NRC dealt with the design basis and license failures with a stroke of a pen in 1992. (See Exhibit L – RCS Operational Leakage)

The table below best provides the chronology as well as the facts, and the implications for the LRA. In simplest terms, Licensee and the Commission, by adopting the GDC defined in Amendment 9 to the original

license application, accepted a draft industry GDC in place of the lawful federally approved General Design Criteria for Indian Point.

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 original 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 was 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 the GDC 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. As noted, "Specifically, for the

Date:	Docketed Activity	Reference	Implications to fidelity of the License Amendment
			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 10 CFR 50. 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 Appendix A.		The commission concurred on January 1982.

Date:	Docketed Activity	Reference	Implications to fidelity of the License Amendment
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 was not required to comply with federally approved General Design Criteria, if construction permits were issued prior to May 2, 1971.** **This is 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 10 CFR 50 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 safety margin and reasonable assurance of protection of the health and safety of the public have been compromised for over three decades, without the public understanding of the loss of margin in safety. Subsequently, Entergy allowed the error to remain and is actually currently committing Unit 2 to trade organization design criteria.**

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 NRC continued the pattern of bypassing the Administrative Procedures Act in 1992 (see exhibit I), when the regulator relieved the Applicant of *all* compliance enforcement to any General Design Criteria, without any attempt to abide by the Administrative Procedures Act.

The issue of whether the Commission could lawfully use guidance documents from trade organizations in lieu of federal rules 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 held 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.

Yet the NRC continues to disregard the express rules set forth in the Administrative Procedures Act.

In summary, Licensee was obligated to meet the requirements of the General Design Criteria as published on July 11, 1967. Indian Point (i.e., *both* the Indian Point 2 and Indian Point 3 facilities) was designed, constructed and is now being operated on the basis of the proposed General Design Criteria, published July 11, 1967. Applicant makes the representation that Indian Point is in compliance on page 3 of the LRA. However, construction of the plant was already underway when (1) the Final Facility Description and Safety Analysis Report was filed on December 4, 1970, (2) when the Commission published its revised General Design Criteria in February 1971, and (3) when the final version of the General Design Criteria was published in July 1971, which included the inaccurate statement, "[W]e 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 [is] now in effect and we have concluded that the plant design conforms to the intent of these newer criteria."

In fact, Indian Point was not in compliance with 10 CFR 50 Appendix A then, and is not in compliance with 10 CFR5 0 Appendix A now. (See current 2006 Unit 2 UFSAR submitted as a part of the LRA.)

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 Commission's interpretation of the meaning of the requirements of the 1971 GDC. 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 because it incorporated by reference regulations in violation of 10 CFR 50.21[2], (regarding equipment aging program scope) and employs a methodology under NUREGS prepared and promulgated outside rulemaking

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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]

procedures. Industry trade guidelines, such as NEI 95-10 Rev. 6, have no legal force. Neither public involvement nor the most fundamental steps required under the Administrative Procedures Act were adhered to by either Applicant or the Commission.

[1] (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;

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(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.

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[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 . 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).* Stakeholders aver 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.

Entergy's Aging Management Plan Lacks Requisite Specifics, and is in large measure a meaningless "agreement to agree" (Contention #11 is fully reference and incorporate as set forth herein).

The 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 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.

Applicant included the NUREG language in the FSAR, but fails to state with specificity how it plans compliance (in this example with General Design Criteria 1971). Setting forth a recitation of regulatory language is not the same thing as demonstrating compliance with regulations.

Regardless, Applicant may clearly not use the Aging Management Program to argue compliance with regulations it is in breach of. Applicant is, in effect, using promises of remedying problems as a means of holding open options that should be eliminated under the Aging Management Rule.

A glaring 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)

The IP2 Final Safety Analysis Report (FSAR) does not address Criterion 35 at all. 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 10 CFR 50 Appendix A.

A detailed list of specific violations contained within 10 CFR Part 54 will be provided in supplemental submittal to this contention. (CONTENTION # 51 of this Petition is referenced and incorporated fully, as if set forth herein.) An example is rovided below from review of the

limited material available to Stakeholders by the Applicant , 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.**

[NOTE: All emphasis in these contentions is added by Stakeholders.]

These additions provide abundant latitude to the Applicant as to what the Architects and Engineers need to do in order to minimally satisfy the criteria **but completely abrogate the public's right to a 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 indicates: "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" add a qualification that renders the provision all but meaningless.** (see Exhibit N page 14).

Applicant bypasses the rules by failing to properly examine or replace reactor core internal components that have known susceptibility to failure, and have, in fact, demonstrated such failure on multiple occasions. For example, the components 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 manifestly 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). This is a patent example of financial considerations trumping public health and safety (see exhibit P and declaration of Ulrich Witte, Exhibit Q1) and constitutes a *prima facie* violation of 10 CFR 50 Appendix A.

Applicant attempts to paint over 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

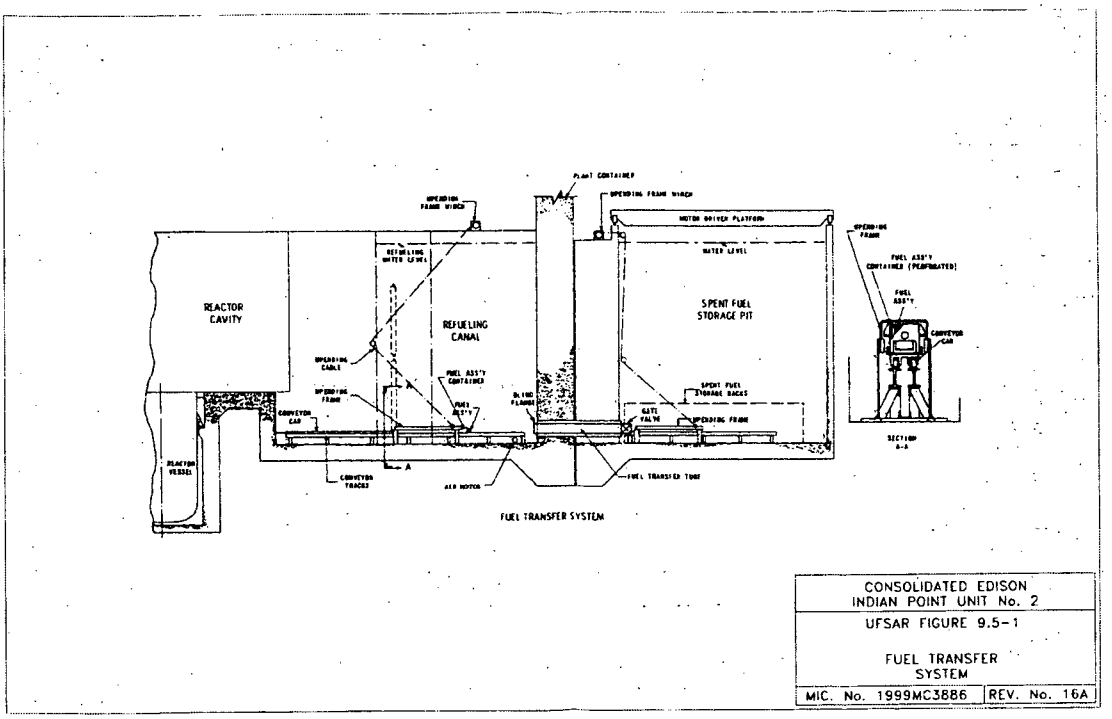
than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval." See section A.2.1.141 of the LRA report.

This language essentially removes this entire matter from the public's right of input and participation, bypassing the procedures required by law in the Administrative Procedures Act. It is further another example of a worthless "agreement to agree."

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 so routine, and the violations so acceptable, that the NRC just recently published a notice regarding a leaking and aging 20-inch pipe, described as a "conduit" with a pinhole leak.



This course undermines the primary mandate of the NRC to protect public health and safety. And Applicant's failure to comply with the specific rules of 10 CFR 54 violates the Administrative Procedures Act.

An example of how this plays out may be found in the 20-inch "conduit" not being considered part of the Aging Management Program or part of the environmental program, ergo, Entergy's failure to engage in inadequate inspection and maintenance of this particular component is not considered unlawful. See exhibit R, Stakeholders ask that this be considered.

The breadth and depth of the issues raised in these contentions are considerable. 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 reckless course chosen by Applicant and the regulatory failure raise questions about any statement made in the LRA, or the Current Licensing Basis for Unit 2 and Unit 3.

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 and Indian Point 3 are clearly analogous, and a new superseding license should, without question, be denied.

For those issues raised here, Stakeholders also aver that the NRC administrative process presents a questionable forum for the adjudication of the NRC's regulatory failure and unlawful acts. Stakeholders question 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 course of conduct.

CONTENTION # 27: The LRA for Indian Point 2 & Indian Point 3 is insufficient in managing the environmental Equipment Qualification required by federal rules mandated that are required to mitigate numerous design basis accidents to avoid a reactor core melt.

Issue Statement: Stakeholder's contend that Applicant's LRA for Indian Point 2 and Indian Point 3 be denied because the proposed LRA is insufficient to demonstrate compliance with either 10 CFR50.49 (e)(5) or 10 CFR54. And 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.

Summary of Contention

Indian Point 2 and 3'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). Entergy's claims credit in their LRA under Table 3.6.1, and EQ analysis in section 4.4, however it is 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 the extinguishers 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 in the past 40 years. 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, and the fire extinguishers were quietly thrown out as each one broke etc. At Indian Point 2 & Indian Point 3, 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.

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

d. 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.

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 & Indian Point 3's LRA Against the Requirement of 10 CFR 50.49

(i) The Indian Point application for Unit 2 and Unit 3 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.

These issues of limited functionality and integrity of certain components and failures, including but not limited to, Instrumentation & Control (I&C) cables, insulation materials for a 60 year period are in contradiction to ACRS recommendations regarding compliance to 10CFR50.49, and the implications to 10CFR50.4. Therefore, Stakeholders contend that the NRC's acceptance of Entergy's LRA is in violation of the Administrative Procedures Act (Exhibit FF).

(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 CC 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[3]

[3] ACRS letter dated June 17, 2002

A discussion of the treatment of the Instrumentation and Control (I&C) cables during the proposed new superceding 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, but not for 60 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.

(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 Loss of Coolant Accident (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**. (emphasis added) 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.

(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 GG1).

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 & Indian Point 3 license renewal as well as IP2 and 3'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 & IP3 –which are presently unqualified and will apparently remain unqualified from Entergy statements in their LRA. Stakeholders assert that Entergy's LRA fails to comply with the 10CFR50.4; (2), the NRC, accepted the unqualified components as acceptable, based upon industry guidance, which violate the legal regulations; (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 violation 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 noticed the approach taken by the NRC and Entergy on other issues, yet Entergy failed to act to comply with the regulations, in particular, with respect to Indian Point 2 and Indian Point 3.

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 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%. (CONTENTIONS 4,5,6,7,8,9,10 are reference and incorporate in full as if set forth herein).

Yet they closed the issue regardless under a high school quality economic analysis.

The approach was not only unlawful but also, technically irresponsible and in violation of the Administrative Procedures Act. 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, Stakeholders contend that the NRC must deny the Applicant's LRA for IP2 and IP3, 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.

Contention #28 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. (Exhibit JJJ)

On January 7, 2007 a sincerely concerned Entergy Nuclear Northeast employee, and the former Quality Control Staff wrote Riverkeeper and expressed these serious issues with regard to reduction in Quality Control (QC) and violations of 10 CFR 50 Appendix B requiring "independent inspections".

" A plan that Entergy Nuclear Northeast has implemented at their Northeast plants TOTALLY ELIMINATES the quality control departments (day to day oversight). ....upper management has dissolved the QC Departments at each of their plants, and is now having each department perform their own inspections!!! For example, Bill, Joe and Sam are maintenance mechanics. On one job Bill and Joe perform the work while Sam inspects the work. On the next job, Bill and Sam do the job, while Joe performs the inspections..... With more than 25 years of quality control experience, we have many examples .... that this was a poor idea. ... Mechanic ague that an unacceptable condition was "close enough", that by spending the TIME to rework the part won't make it that much better. Now that mechanics inspect the work of their peers, we now say amongst ourselves that "close enough", has now become "good enough".

Such logic would be like the State saying that since NYS citizens are inherently honest, and to save a substantial amount of money on police patrols, the State was going to make drivers and passengers accountable for the driver's driving. When a passenger identified a driver violating a traffic law, they would report the violation. That logic would be simply laughable, except that is what Entergy Nuclear Northeast has done to quality oversight at its Northeast Plants."

This serious reduction in QC directly impacts the proposed 20 year new superceding license. The current condition of the plant will continue to decline under this new system. The overall nuclear safety culture trends from 1999-2006 indicates that since 2002 the overall trend in Nuclear Safety Culture has been declining since 2002. Indian Pont has shown a sharper decline than other Entergy fleet plants. Additionally the implications of carrying over this reduced QC program into the new proposed 20 year license period must be fully disclosed and evaluated with regard to an aging management plan.

Stakeholders contend that 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.

CONTENTION 29: (CONTENTION # 28 of this Petition is referenced and incorporated fully, as if set forth herein.) 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 there are 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 2nd quarter 2007. (exhibit KK)

CONTENTION 30: (CONTENTION # 28 of this Petition is referenced and incorporated fully, as if set forth herein.)A second example is Entergy's ineffective quality assurance program did not catch a trend of deficient procedures associated with temporary modifications. 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.*

CONTENTION 31: (CONTENTION # 28 of this Petition is referenced and incorporated fully, as if set forth herein.) 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.

CONTENTION 32: (CONTENTION # 28 of this Petition is referenced and incorporated fully, as if set forth herein.) 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.

CONTENTION #33: The EIS Supplemental Site Specific Report of the LRA is misleading and incomplete because it fails to include refurbishment plans meeting the mandates of NEPA, 10 CFR 51.53

post-construction environmental reports and of 10 CFR 51.21.
Issue Summary.

Stakeholders contend that the Applicant's LRA is misleading and incomplete because Entergy inaccurately alleges in its EIS Site Specific Report, marked as Appendix E to the LRA, that there are no refurbishment issues anticipated during the new superseding license period, and therefore that no environmental costs need be considered.

However, Entergy has already prepared for a major refurbishment that may have significant environmental impacts and cost by ordering Replacement Reactor Vessel Heads for both Indian Point 2 and 3. In fact an order form shows that preliminary delivery dates have been scheduled.

Stakeholders additionally assert that the Applicant's LRA fails to comply with 10 CFR 51.21 and 10 CFR51. 23, by failing to provide a refurbishment aging management plan, for already planned refurbishment during the proposed 20 year new superseding license.

Legal Basis: Applicant is required, in the EIS Supplemental Site Specific Report of the LRA, to fulfill the requirements of NEPA. The rules codified in 10 CFR 51.21 and 51.53 require NRC licensees filing a LRA to include, as a part of the EIS Supplemental Site Specific Report, any refurbishment

issues or plans and to lay out the environmental risks and costs associated with refurbishment. Applicant has failed to comply with this rule.

Facts and Issues in Contention Entergy's Representation

In the LRA for Indian Point 2 and 3, in Appendix E of the EIS Statement (Supplemental Environmental Report, section 3.3 of the Environmental Report Refurbishment Activities), Applicant states that "there are no such refurbishment activities planned and/or anticipated at this time" and therefore that no environmental costs need be considered. Below are Entergy's claim that **No Refurbishment Activities have been Identified** and therefore there are no environmental concerns or mitigation is required to be addressed in the LRA. As specifically set forth in Appendix E of the EIS, section 3.3., sections A,B,C,D,E,F, H, I:

> A. Refurbishment impacts on terrestrial resources [10 CFR 51.53(c)(3)(ii)(E)] NONE. **No refurbishment activities have been identified**. Consideration of mitigation is not required.
>
> B. Threatened or Endangered Species (for all plants) Threatened or endangered species [10 CFR 51.53(c)(3)(ii)(E)] SMALL. **No refurbishment activities have been identified.**
>
> C. Air Quality Air quality during refurbishment [10 CFR 51.53(c)(3)(ii)(F)] NONE. **No refurbishment activities have been identified.** Consideration of mitigation is not required.

D. Socioeconomics Housing impacts [10 CFR 51.53(c)(3)(ii)(I)] SMALL. **No refurbishment activities have been identified.**

E. Unavoidable Adverse Impacts 6.3.1 Requirement [10 CFR 51.45(b)(2)] The applicant's report shall discuss any adverse environmental effects which cannot be avoided upon implementation of the proposed project. Public utilities: public water supply availability [10 CFR 51.53(c)(3)(ii)(I)] SMALL. **No refurbishment activities have been identified.**

F. Education impacts from refurbishment [10 CFR 51.53(c)(3)(ii)(I)] NONE. **No refurbishment activities have been identified.** Consideration of mitigation is not required. Offsite land use (effects of refurbishment activities) [10 CFR 51.53(c)(3)(ii)(I)]

H. Historic and archaeological properties [10 CFR 51.53(c)(3)(ii)(K)] SMALL. **No refurbishment activities have been identified** and no increases in total number of employees during the period of extended operation are expected. Further consideration of mitigation measures is not warranted. Historic and archaeological properties [10 CFR 51.53(c)(3)(ii)(K)]

I. Offsite land use (effects of license renewal)[10 CFR 51.53(c)(3)(ii)(I) NONE. **No refurbishment activities have been identified**. Consideration of mitigation is not required. Offsite land use (effects of license renewal) [10 CFR 51.53(c)(3)(ii)(I)].

The NRC contemplates and anticipates that during the relicensing period of any nuclear plant substantial refurbishment will be required to maintain the plant in compliance to regulations. The claim that "no refurbishment activities have been identified is a contradiction to the realities of any aging facility and indicates Entergy's attempt to omit relevant information in the LRA which directly impacts the aging management plans.

The Replacement Reactor Vessel Head Order

However, the Applicant has, in fact, already prepared for a major refurbishment by ordering Replacement Reactor Vessel Heads for Indian Point 2 and 3, with delivery dates scheduled for October 2011 and October 2012, respectively. This refurbishment is evidenced by a presentation made by the Doosan Heavy Industries Construction Co., Ltd presentation at the Burns & Roe 17th Annual Seminar, *Powering the Future*, on March 21, 2007 (Exhibit DD)

The order states:

> **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)

Entergy only purchased these heads for Indian Point and two other facilities; not for its entire fleet.) (See Exhibit DD, A true and accurate copy of the PDF web based file is submitted as proof.)

The Doosan document evidences that Applicant already has plans for refurbishment during the proposed 20 year new license period.

Stakeholders assert that Applicant has a regulatory obligation to disclose these plans, even if modification and installation dates are not established.

Misleading Nature of Entergy's EIS

Entergy is a multinational corporation with extensive knowledge and expertise in the nuclear reactor industry and with ownership rights to eleven nuclear reactors in America. Entergy also offers its services as a supplier of expert assistance in the filing of LRAs to other NRC licensees considering a 20 year license renewal for their own facilities.

Refurbishment on the scale of a reactor head replacement, which has already been ordered, and with specific delivery dates, indicate that the omission by the Applicant was not an oversight.

Substantial engineering and construction work is required for this material refurbishment

Since 2003, there have been degradation issues of rust in the dome and boric acid corrosion in the reactor vessel head at Indian Point. This is a potentially serious problem which mandates remedy.

The reactor head and CRDMs purchase is costly, approximately 15 - 20 million dollars per unit.

There are numerous issues that will affect land use. Primary among them, is the lack of an off site repository for Indian Point's radiological and mixed waste streams.

Mitigation Issues

Postulated Accidents: Severe accident mitigation alternatives [10 CFR 51.53(c)(3)(ii)(L)] SMALL. No impact from continued operation. There are a host of accident scenarios that must be evaluated, with various mitigation alternatives explored.

In the specific area of reactor vessel head replacement and repair, the installation of a reactor vessel head is a highly complex operation which comes with a host of potentially significant issues, including the possibility of cutting a hole in the containment.

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 incomplete and misleading LRA, in a violation of 10 CFR50.5 and 10 CFR 50.9, by attempting to hide significant environmental, health and safety concerns in an attempt to streamline approval of the LRA, even though the omission of refurbishment planned

during the proposed 20 year new license, 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' 10 CFR 53(c) (1)(2). Entergy did not provide a necessary comprehensive EIS, as required under section 102(C) of the NEPA. From 10 CFR 50.59:

> 2) A licensee shall obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:
> (i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);
> (ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);
> (iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);
> (iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);
> (v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);
> (vi) Create a possibility for a malfunction of an SSC important

to safety with a different result than any previously evaluated in the final safety analysis report (as updated);
(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or
(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

Further, 10 CFR 50.59 (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. Applicant shall assess the impact of the proposed action on threatened or endangered species in accordance with the Endangered Species Act'.

(c) Entergy applies a cut-and-paste EIS to Indian Point that is not rightfully characterized as a SEIS, and Entergy, as a Fleet Operator, has claimed no refurbishment issues exist at any of its reactor sites

Regardless of the distinct site specific characteristics at its nuclear plants, no refurbishment issues seem to ever crop up that Entergy deems worthy of SEIS analysis.

Entergy appears to use the same sort of non-site specific, generic environmental analysis for Indian Point that it applies to other reactors in its fleet. Indeed, a review of Environmental Reports filed at six other Entergy

plants, reveals the virtual verbatim denial of refurbishment issues that Entergy employs with respect to Indian Point 2 & 3. To wit:

> Arkansas One Plant (ANO-1): "[A]n evaluation of structures and components as required by 10 CFR 54.21, **did not identify any major plant refurbishment** activities or modifications necessary to support the continued operation of ANO-1 during the license renewal term. Therefore, evaluation of refurbishment issues was not considered."
>
> Wolf Creek (WCGS): "WCGS has stated that its evaluation of structures and components, as required by 10 CFR 54.21, **did not identify any major plant refurbishment** activities or modifications as necessary to support the continued operation of WCGS for the license renewal period. In addition, any replacement of components or additional inspection activities are within the bounds of normal plant operation, and are not expected to affect the environment outside of the bounds of the plant operations evaluated in the U.S. Nuclear Regulatory Commission's 1982 Final Environmental Statement Related to Operation of Wolf Creek Generating Station, Unit No. 1".
>
> Vermont Yankee (VYNPS): "VYNPS has stated that its evaluation of structures and components, as required by 10 CFR 54.21, **did not identify any major plant refurbishment activities or modifications** as necessary to support the continued operation of VYNPS for the license renewal period. In addition, any replacement of components or additional inspection activities are within the bounds of normal plant operation and are not expected to affect the environment outside of the bounds of the plant operations evaluated in the U.S. Atomic Energy Commission's 1972 Final Environmental Statement."
>
> Pilgrim (PNPS): "[PNPS's] evaluation of structures and components, as required by 10 CFR 54.21, **did not identify any major plant refurbishment activities or modifications as necessary** to support the continued operation of PNPS for the license renewal period. In addition, any replacement of

components or additional inspection activities are within the bounds of normal plant operation, and are not expected to affect the environment outside of the bounds of the plant operations evaluated in the U.S. Atomic Energy Commission's 1972 Final Environmental Statement Related to Operation of PNPS."

Nine Mile Point's (NMPN): "[NMPNS's] evaluation of structures and components, as required by 10 CFR 54.21, **did not identify any major plant refurbishment activities or modifications** as necessary to support the continued operation of NMP, for the license renewal period. In addition, any replacement of components or additional inspection activities are within the bounds of normal plant operation, and are not expected to affect the environment outside of the bounds of the plant operations evaluated in the U.S. Atomic Energy Commission's 1972 Final Environmental Statement Related to Operation of Nine Mile Point Nuclear Station."

Brunswick (BSEP): "[BSEP's] evaluation of structures and components, as required by 10 CFR 54.21, **did not identify any major plant refurbishment activities or modifications** as being necessary to support the continued operation of BSEP for the license renewal term. In addition, any replacement of components or additional inspection activities that are within the bounds of normal plant operation are not expected to affect the environment outside the bounds of the plant operations evaluated in the Final Environmental Statement Related to Operation of Brunswick Nuclear Steam Electric Plant Units 1 and 2, issued by the U.S. Atomic Energy Commission in 1974."

It is exceptionally hard to believe that no refurbishment will be needed during the next 20 years at all 7 of these nuclear facilities. Nuclear plants are made of concrete and steel, moving parts, endless piping, and highly corrosive radioactive nucliedes. The NRC has anticipated

refurbishment would be necessary for most, if not all, plants who are looking to extend their lifetime by 20 years.

It is also unfathomable that the NRC has accepted so many Entergy LRAs that have not identified **any major plant refurbishment** activities or modifications necessary to support the continued operation during the various proposed license renewal terms of 20 years.

(d) Replacement of a reactor vessel head for Indian Point 2 & 3 are not only refurbishment issues, but are a significant environmental issue that affect public health and safety. The means and method of disposal of the irradiated old reactor vessel heads must be addressed, in the Aging Management Plan. Indian Point 1, 2, or 3 were neither designed nor licensed to act as radioactive waste storage facilities.

With the closing of Barnwell to Indian Point's low level radioactive waste as of 2008, the impacts of any and all radioactive waste streams generated at Indian Point 1, 2 & 3, including disposal of old reactor vessel heads, are an issue of paramount importance for the safety the Stakeholder community (CONTENTION #43 of this Petition is referenced and incorporated fully, as if set forth herein).

(e) Applicant has failed to provide the requisite specificity for mandated reports and has also failed to provide required environmental reports for its plans to change, modify or refurbish the facility.

(f) Stakeholders contend that any reactor refurbishment issues contribute to potential environment, health and safety risks. Hiding or ignoring significant information is contrary to the NRC regulations requiring LRAs to be complete, accurate and truthful. The NRC was put on notice, prior to the NRC's acceptance of the LRA for review that it was incomplete and inaccurate. The NRC responsibility under the Administrative Codes Act is to enforce its own regulations.

i. **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 & 3 is within scope. Therefore, this contention brought by the Stakeholders against Indian Point 2 & 3 regarding refurbishment is within the scope of Entergy's LRA.

ii. Contention Raises Material Issues of Fact and Law

Issues of fact and law are present in this contention. Reactor vessel

head replacement is never a like-for-like switch of components or equipment. It is one of the most critical refurbishments that a reactor operator can undertake. In some cases replacement of the reactor vessel head can require cutting a hole into the containment.

1. Reactor vessels are not tangential components. They contain the nuclear fuel in the plants and, over time, are irradiated which can lead to embrittlement, deterioration, loss of material, and degraded ability 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 the vast pool of available knowledge, Applicant neglected to list, describe or report the vessel head replacement, or any other proposed or possible refurbishment actions during the entire 20 year new license period, in the LRA's EIS, Appendix E.

2. The omission of significant refurbishment issues from the EIS Appendix E is an egregious 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 10 CFR 53 and 10 CFR 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. 10 CFR 53 and 10 CFR 54.21 require 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.

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 hundreds of workers, including senior members of management. Omission of such a significant project from the LRA application for Indian Point 2 & 3 constitutes serious violations 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.*

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.

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. Under the regulations licensees who cannot achieve and maintain adequate levels of safety are not permitted to continue to conduct licensed activities to protect public health and safety.

Contention is Supported By Facts and Expert Opinion

This contention is supported by the fact that the Entergy's placed orders with Doosan for new reactor vessel heads for IP 2 and IP3 to be delivered for use during the new 20 year license period, yet Applicant failed to include this pertinent information in its LRA.

Herein, the Stakeholders raise issues of both fact and law. Entergy, at best, has made a critical error which should cause the NRC to dismiss the LRA. At worst, Entergy has purposely omitted facts, thereby misrepresenting its plans to the NRC and the public for the proposed 20 year new superseding license. The undersigned Stakeholders therefore

respectfully request that the Applicant's LRA be denied due to the fatal errors.

The Stakeholders contend that as a matter of law and fact the Applicant's LRA fails to include a site specific EIS, and therefore must be rejected as incomplete and inaccurate. Additionally Stakeholders contend that the integrity of the NRC's entire EIS scoping process and evaluation must be further questioned, due to it repeated failure to require site specific EISs.

CONTENTION #34: Stakeholders contend that accidents involving the breakdown of certain in scope parts, components and systems are not adequately addressed Entergy's LRA for Indian Point 2 and Indian Point 3.

Issue Statement: Stakeholders contend that accidents involving the breakdown of certain in scope parts, components and systems are not adequately addressed Entergy's LRA for Indian Point 2 and Indian Point 3.

Accidents involving the breakdown of certain in scope parts, components and systems present greatly increased risks of an offsite migration of radiological contaminants. These kinds of accident scenarios, and their costs and impacts must be included in the Indian Point 2

& 3 LRA for the proposed 20 year new superseding license. However they are not.

Aging Management issues that should be reviewed include, but are not limited to, the following:

a) Boric acid corrosion (BAC) represents a significant aging management issue affecting primary systems at Indian Point 2 & 3 which could lead to release of radioactive contaminants into the environment. Aging management of BAC must be fully addressed in the Applicant's LRA. It is not. The Aging Management in the LRA plan for this important issue fails to adequately address, for example, valve packing and valve body-to-bonnet gaskets. The fact that IP2 and IP3 are already working on the engineering difficulties involved in a complicated and dangerous reactor vessel head replacement is a significant issue that can result in an accidental release of radioactivity into the environment from reactor vessel head failure. Such accident pathways must be included in the LRA.

b) Aging management of reactor vessel internals bolting must be fully addressed in the Applicant's LRA. It is not. The reactor vessel internals bolting at Indian Point 2 and 3 are susceptible to age-related degradation which could lead to an off site release of radioactive contaminants. The LRA and UFSAR documents fail to lay out an adequate

aging management plan for inspection and replacement of reactor vessel internal baffle bolts.

c) Aging management of the fuel rod control system must be fully addressed in the Applicant's LRA. It is not. There are serious safety and environmental concerns related to Indian Point's inadequate Aging Management Plans for the Fuel Rod Control System, that includes dropped rod events, unplanned plant trips, complete equipment failure, shut-downs, and highly dangerous at-power-maintenance attempts.

d) Aging management regarding Severe Duty Valve failure must be fully addressed in the Applicant's LRA. It is not. Severe Duty Valve failure, further complicated with sourcing issues for many approved valves which are no longer available, create serious potential risks to Indian Point's ability to accomplish and maintain a safe shutdown of the facility. These valves include, but are not limited to: Feedpump recirculation control valves; Feedwater regulating valves; Atmospheric dump valves; Condenser dump valves; Feedpump discharge check valves; Feedpump discharge check valves; and Pressurizer spray valves. Failure of these valves, or inability to find and obtain approved replacement valves, directly impacts safety and reliability of the plant during the 20 years of the new superseding license period.

e) Aging management of the briny reactor water coolant environment with regard to microbial corrosion of the piping and with regard to zebra mussels, must be fully addressed in the Applicant's LRA. They are not. The reactor water coolant environment can have dramatic negative effects and increase the fatigue on important pressure water components, and greatly increase pipe leakage, which, in turn, can lead to significant pipe burst events or core damage events. Issues of the briny water used as reactor water coolant must be fully addressed in the LRA.

f) Aging management of cable degradation, especially in underground wet circuits must be fully addressed in the Applicant's LRA. It is not. Cable degradation, especially in underground wet circuits, is a pathway to massive circuit failures that lead to loss of the operator's ability to safely shut down the reactors. Further, these wet circuits, and generally known fatigue issues surrounding medium voltage Ethylene Propylene Rubber Cables, could create a serious electrical fire, as the cables can reach a point of electrical breakdown. The NRC has raised concerns on this very issue.

g) The cumulative effect of this constant exposure of the reactor vessel to neutron irradiation and the aging management plans to mitigate severe reduction in the fracture toughness and ductility of the PWR internal

must be fully addressed in the Applicant's LRA. It is not. The reactor vessel internals have been, and continue to be, exposed to neutron irradiation which in turn causes a severe reduction in the fracture toughness and ductility of the PWR internals.

h) Aging management of refurbishment issues must be fully addressed in the Applicant's LRA. They are not. Entergy alleges there are no refurbishment issues to be considered in the LRA. However, there is a far greater than 50 percent chance that Indian Point 2 and 3 are facing the necessity of replacing feedwater heaters. Lack of industry expertise, fewer vendors and manufacturers, combined with the need for material changes, are serious issues that negatively impinge on the Applicant's ability to maintain safe operation of the reactors.

i) Aging management of Primary Water Stress Corrosion Cracking (PWSCC) must be fully addressed in the Applicant's LRA. It is not. PWSCC appears in heat affected zones of the stub runner/divider plate weld, not mentioned in Entergy's LRA Appendix E, can result in significant impacts.

j) Aging management specifically of PWSCC of Alloy 600 and its weld metals must be fully addressed in the LRA. It is not. PWSCC of Alloy 600 and its weld metals is a serious issue which impinges on both

upper and lower reactor pressure vessel head penetrations. Additionally, this issue potentially manifests itself in reactor coolant system piping, lower head pressurizer penetrations and other components at Indian Point. Ongoing weld failures, a serious shortfall in technology keeping up with site degradation, combined with fatigue, make this a potentially significant pathway for environmental contaminations and or accident pathways.

k) Fatigue of metal components void swelling of reactor internals, as well as serious issues regarding Entergy's inability to visually examine certain difficult, if not impossible, to reach components and containments creates pathways resulting in significant release accidents.

l) Aging management of shell and heat exchanger replacement must be fully addressed in the Applicant's LRA. It is not. Shell and heat exchanger replacement will inevitably occur during the 20 year new superseding license.

m) Appendix E of Entergy's LRA fails to address any accident analysis for events that are beyond the current design basis for Indian Point 2 & 3.

n) Entergy's LRA Environmental Supplement fails to address the obsolescence concerns as relates to digital upgrade of the rod control logic and power cabinets at Indian Point 2 & 3.

o) Entergy's LRA Environmental Supplement fails to address the risks associated with lo-temperature flow-accelerated corrosion (FAC), including unanticipated emergency shutdowns.

p) Entergy's LRA Environmental Supplement fails to address the known industry wide problem of securing and having on hand contingency spare parts.

q) Entergy's LRA Environmental Supplement to the GEIS, fails to address the shortage of seasoned engineers with the knowledge pool to maintain the aging Indian Point Reactors. This severe intellectual shortage becomes crucial in numerous cases, such as where reverse engineering would be necessary to build replacement parts which are no longer available on the open market. Even if reverse engineering is possible, the replacement parts would no longer be a like-for-like replacement.

r) Entergy's LRA Environmental Supplement Appendix E fails to adequately address known premature failure of containment coatings, resulting in significant impacts and costs of such accident pathways

s) Entergy's LRA Environmental Supplement fails to address the industry wide, and site specific, problem of ever increasing obsolescence issues with original equipment installed for Indian Point's instrumentation, control and safety system applications.

t) Entergy fails to adequately address critical Reactor Pressure Vessel (RPV) issues in the LRA, the UFSAR, and in the Appendix E EIS Supplemental Report. The RPV is the critical component for plant life management, due to the unacceptable consequences of its failure and due to the difficulty of its replacement. The RPV is subjected to neutron irradiation in the core region, which results in irradiation-induced embrittlement that may lead to a shift of the ductile-to-brittle transition temperature. Notably, both industry and NRC have admitted to a severe lack of knowledge in this area.

u) Cables are critical for plant safety and operation and shutdown at Indian Point 2 and Indian Point 3, yet Entergy fails to present an adequate aging management program for these critical components. Degradation of these cables could lead to accidents at the site resulting in an electric fire destroying major plant components and infrastructure, including, but not limited to, key safe components necessary for safe shutdown, that would in turn lead to core meltdown.

Conclusion: Based on all of the above, Stakeholders submit there are apparent and significant fact issues that mandate analysis. The failure of Applicant to address these and other critical issues is fatal to the LRA.

CONTENTION 35: Leak-Before-Break analysis is unreliable for welds associated with high energy line piping containing certain alloys at Indian Point 2 & Indian & Indian Pont 3.

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

The Leak-Before-Break 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, 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 concept means that the failure mode of cracked piping is a leaking through-wall crack, which, it is assumed (1) may be timely and safely detected by the available monitoring systems, and (2) does not challenge the pipe's capability to withstand any design loading. The assumptions underlying LBB are based on experience that

double ended breaks and other catastrophic failures of primary circuit piping are unlikely. Various design, operation, inspection and monitoring aspects have been considered as prerequisites.

In recent years and months, Indian Point 2 & 3 have had a disturbing track record of pipe integrity problems, as noted in the below time line taken from reports in the area's paper of record, the Journal News:

> September 20, 2005: NRC and Entergy notified 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 Indian Point 1 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 non-radioactive 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 non-radioactive 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, 2007 (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 reported to have been found. (Stakeholders believe it is actually leak in a 20-24 inch fuel transfer pipe, that is leaking radioactive effluent).

One prerequisite is that locations of piping systems that are susceptible to stress corrosion cracking and 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, such as at VC Summers and other PWR plants, where these welds have made use of Leak-Before-Break (LBB) for questionable weld alloys.

Industry guidance, as well as emerging regulatory funded studies, memorialized in NUREG "Conference on Vessel Penetration Inspection, Crack Growth and Repair," have specifically warned against traditional reliance upon LBB credited in Section 4.7.2 of IP2 and IP3 Section 4 LRA, in spite of the nickel-based alloy weld. [page 4.7-2 of the LRA].

The LRA for Indian Point 2 & 3 does not respond to the warning regarding this potential safety threat, and relies wholly on previously out-

dated studies such as WCAP-10977m and WCAP-10931. 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, that the licensees of 40 pressurized water reactors will raise levels of vigilance concerning reactor coolant system (RCS) welds. The Commission 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 NRC advised that measures be put in place and welds inspected during an outage before the end of 2007. If no outage is scheduled this year, plant operators must justify an extended schedule to the NRC.

Stakeholders' concerns center 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 CALs are merely 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 II2.

Therefore Stakeholders reiterate that the NRC must deny the 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, putting at risk public health and safety during the 20 year new superseding license.

CONTENTION #36: 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 the LRA for Indian Point 2 and Indian Point 3 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) through 60 year old piping, during the proposed 20 year new license. 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. Stakeholders submit the Declaration of Mr. Ulrich Witte in support of this contention.

In Applicant's LRA, the 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. The dialogue is between the ACRS and Entergy and concerns the Extraction Steam System. Entergy's admissions on the weakness of the reliability of the methodology are central to this issue.

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 inspection frequency is the key to a valid FAC management program. And it is *precisely* the accurate specification of inspection frequency which Entergy admits potentially contains enormous errors.

Entergy proposes, via 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 outlines the scope of the FAC program by inference and directly from the LRA, only to include limited piping within 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 & Indian Point 3 have had recent increases its operating power levels, IP3 received an uprate (stretch)of 4.85% on 4/15/2005 and IP2 received an uprate (recal of instrumentation uncertainties) of 1.4% on 5/22/03, CHECWORKS will not be useful.

The profiles required for CHECWORKS and the grid check points are unsubstantiated as a result of these significant changes. The CHECWORKS model cannot be used to determine inspection frequency at Indian Point 2 &3.

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 only so long as plant parameters, such as velocity and coolant chemistry, do not change markedly. 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 & 3.

Indian Point has a track record of broken pipes due to corrosion. The steam generator failure is a design basis accident with very low Probabilistic Risk Analysis (PRA) prediction rate. Thus, 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 precedent regarding LRA acceptance, and Safety Evaluation Report (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 10 CFR 54.21(a)(3) and 10 CFR 50.

CONTENTION 37: <u>The LRA and the UFSAR's for Indian Point inadequately address the currently existing (known and unknown) environmental affects and aging degradation issues of ongoing leaks, and fail to lay out workable aging management plans for leaks and critical safety systems</u>

<u>Issue Statement</u>: Stakeholders assert that the LRA for Indian Point 2 & 3 fails to lay out, in detail, a workable aging management plan to deal with accidental leaks in the underground pipes, steam pipes and other systems critical to safe shutdown and cooling of the reactor, and cooling of the spent fuel pools. The LRA and the UFSAR's for IP2 & IP3 inadequately address the currently existing environmental effects (both known and unknown) of recent and ongoing leaks, and fails present a workable aging management plan for leaks, including but not limited to, a plan for locating, stopping and remediating current and future leaks during the proposed 20 year new license period.

Any unmonitored releases are in violation of the NRC regulations §20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that -

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem

(1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003 NRC's Regulations.
§ 20.1302 Compliance with dose limits for individual members of the public.

(b) The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in § 20.1301.

Indian Point units 1, 2 and 3 are currently in violation of these regulations and the LRA for IP 2 and IP 3 does not present an aging management plan that adequately addresses leak issues.

Over the past few years, and upon information and belief, currently, unplanned, unmonitored leaks of liquid radioactive effluents, including tritium, strontium 90 and cesium 137, have leaked and are leaking from Indian Point into the groundwater and Hudson River ("Radiation Leaks"). In most cases, the duration, sources, extent, and flow paths 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

manifestly 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 a multitude of and separate 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.

References to leaks in the LRA are poorly stated, vague and ambiguous include but are not limited to, the following:

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. The LRA for IP2 & IP 3 fails to provide adequate proof of a proper safety analysis of this critical seal. The LRA also fails to provide a

detailed aging management plan, despite industry knowledge of leakage associated with this critical component. Unexpected and abnormal shaft movement or misalignment can introduce motions, including but not limited to, shaft tilt, radial offset and orbit, depending on the magnitude and scope of this displacement. Thus the seal arrangement creates potentially dangerous site specific operational issues of concern, as well as site specific wear (aging) effects that must be accounted for with a detailed site specific aging management plan in the LRA.

2. In the LRA Applicant asserts that the feedwater heater is outside the scope of License Renewal. Stakeholders disagree. The feedwater heater is a crucial component in maintaining thermal performance. More importantly, unchecked aging issues contribute greatly to increased pipe fatigue and failure. Pipe degradation, in turn, creates increased leakage problems 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 estimate the life expectancy of stainless steel pipes to be 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. Indian Point has been in operation now for more than three decades. Yet Entergy's aging management plan provides no detailed explanation regarding the adequate management of chemistry, or fundamental maintained requirements such as those required in 10 CFR 50.65. There are no commitments by the Applicant that provide a viable and workable pipe or component replacement strategy for key component pipes needed for the cooling and safe shutdown of the reactor during the 20 year new superseding license period.

Stainless and carbon alloy pipes are cracking and breaking at Indian Point 2 and Indian Point 3. On September 7, 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. An article in the Journal News reported:

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 1 or 2 inches in diameter. However the pipe in question is actually a 20-24in pipe. Additionally, the leak is more than likely, to be a leak in the fuel transfer tube, which may have a significant impact on the integrity of the facility.

The irradiated water from the Radioactive Leaks and all the other leaks flow into fissures in the bedrock under the plant, and into the groundwater, 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. Therefore the irradiated effluent can take

a very convoluted route into the environment, the groundwater and the Hudson River.

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 or degradation that has not been adequately addressed by the Applicant. Ordinary maintenance failed to reveal the specific locations of numerous Radiation Leaks. Therefore the limited aging management programs indicated in the LRA will assuredly fail to identify and stop radiation leaks before they cause damage to the environment, or before the leaks become breaks.

There is no aging management plan to address known potential pipe bursts in piping adjacent to plugged tubes in IP2 and IP3's LRA. Entergy became the operator of IP2 and IP3 in 2001 and the NRC stated in its report of October 5, 2005, that Applicant had not checked monitoring wells since transfer of ownership. When the monitoring well was finally checked, tritium contamination ten times the EPA drinking water limit was found. This is indicative of Applicant's negligence in conducting routine

inspection. Since Applicant 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, the LRA fails to proposed an adequate aging management program, for an already corroded system of pipes.

The unmonitored, multiple leaks at Indian Point 2 and Indian Point 3 are symptomatic of an aging pipe system, and provide direct evidence of underground pipe failure and/or degradation due to the aging of various systems. Such systems have not been adequately inspected or addressed by Applicant even during the present license term. The proof that Applicant will be properly managing aging issues is wholly inadequate.

On April 7, 2007, Radiation Leaks, including tritium leaks allegedly from underground pipes on the "non-radioactive" side of plant, were discovered purely by random accident, rather than via a coordinated, intelligent aging management and inspection plans. Other leaks were discovered only by accident. In one instance tritium contaminated water was found seeping from surface cracks in spent fuel pool number 2 because of special excavation work being done by an independent contractor. The crucial point is that these discoveries were made either by accident or because of special investigations conducted following such happenstance

discoveries. The Radiation Leaks were not discovered through regular inspection and maintenance. The length of time and extent of the Radiation Leaks have existed remains unknown.

There is no reason to believe that Applicant will do a better job of properly inspecting, maintaining and managing the aging facility during the 20 years of the new superseding license. In fact the LRA does not even identify an aging management plan to locate, stop and remediate the *current* and leaks, much less future leaks. Entergy presents only vague references to best industry standards and sparsely defined sketches of potential aging management plans. Specific and detailed plans are needed to deal with leakage issues caused from corrosion, fatigue, thermal shock, FAC (flow-accelerated corrosion), and other leakage causes of concern during an additional two decades of operation.

Critically, compromised pipes fail due to effects associated with aging – including, embrittlement, corrosion, rust, heat and microbiological and chemical agents, may destabilize and weaken the tensile strength of the pipes and associated equipment and components, causing serious accident pathways, including core damage events. This presents an unacceptable risk during the proposed 20 year extended life of the plant. Pipe integrity must be specifically and fully addressed by the Applicant's aging management

plan set forth in the LRA. However the aging management plan proposed in Entergy's LRA utterly fails to do so.

Aging issues associated with leaking pipes and radioactive effluent include, but are not limited to:

a. Unchecked and unremediated leaks that will further increase the contamination levels in potable water sources, the groundwater, the air (as a consequence of evaporation), and the Hudson River. Radioactive contamination is cumulative, and therefore the proposed increase of an additional 20 years of operation of Indian Point significantly increased the radioactive contamination. The public health risk is greatest to babies in utero, infants, children and women.

b. The risk of wall collapse in one of the spent fuel pools is these known. Non-specifically identified leaks hold the potential to increase the rates of corrosion in the underground pipes and the other structures at the Indian Point site.

c. Significant issues of fact exist, as these unreachable pipes and systems cannot be tested with any certainty. Reactor coolant chemistry is considered a key issue of concern in Flow Accelerate Corrosion. Water chemistry inside the pipes of the reactors is a know concern. It follows that 250,000 gallons of radioactive contaminated

water under the site must also be evaluated as to the corrosion impacts on the outside of the pipes.

Ignorance is not a foundation for sound engineering, especially at a nuclear power plant, and the unknown composition and layout of Indian Point's piping invalidates the LRA,

It is a matter of basic engineering as well as common sense that Entergy can neither devise nor implement an adequate aging management plan without a complete and comprehensive knowledge of the composition and layout of the underground piping system. This gaping hole in Applicant's knowledge about the nuclear facilities it has now run for more than six years was confirmed at an April 26, 2007 public NRC meeting in Cortlandt, N.Y. ("April NRC meeting"). At the April NRC meeting, representatives from both Entergy and the NRC publicly conceded that they did not even know the metallurgic composition of much of the underground piping.

This fact alone should invalidate the LRA and result in the denial of a new superseding license for Indian Point 2 & 3.

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 shutdown of the IP2 and IP3 reactors. The condition of pipes and components in a buried or embedded environment, in particular, can only be guessed at.

Thus, the Applicant cannot assure the NRC and the public that it will be able to manage effects of aging, soil elements, the intake of microbial, brackish water from the Hudson River or storm surges during the proposed 20 year new superseding license period.

The unique chemistry of the Hudson River is of particular import, and is a factor which has been essentially ignored in Applicant's LRA. (As noted in another contention, global warming is another factor relevant to piping and other buried systems which has been ignored.)

The LRA for Indian Point 2 & 3 further fails to address the unique corrosion issues associated with the use of microbial, brackish water in the coolant process

Notably, the Hudson River water and irradiated water has already caused dangerous corrosion of Indian Point's piping, valve and gauge system, as evidenced by the recent and current Radiation Leaks. There are multiple leaks in underground piping throughout the plant. Stakeholders

have seen plume maps shown at a meeting with the NRC and Entergy, that reveal large areas of the plant – including the discharge channel – have been tested and show evidence of underground radioactive effluent. These plume maps and leak reports were due to be submitted by the Applicant to the NRC in the fall of 2007. But, to date, the maps have not been made available to citizen Stakeholders for inspection, review and inclusion in Petitions.

Entergy now asserts that this vital information, which directly affects public health and safety, is proprietary. The NRC's apparent acceptance of Entergy claim of proprietary ownership to this information prevents Stakeholders access to relevant information necessary to write sustainable Contentions.

Therefore Stakeholders expressly reserve the right to amend this Petition after the leak plume maps and leak reports, and the New York DEC independent studies based on captured aquatic life and other tests regarding Essential Fish Habitat or Significant Coastal Fish and Wildlife Habitat, specifically of the Haverstraw Bay, regarding the groundwater contamination are complete and fully reviewed are made available to Stakeholders.

Leak History

To date Applicant still has not identified where the multiple leaks are, how

long they have been leaking, or to what extent is the groundwater contaminated under the plant. The following is a partial list of the ongoing leak problems at Indian Point 2 and Indian Point 3:

> 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 assures the public there is no "immediate risk to public health or the environment." 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 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.
>
> 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. Exposure to tritium increases the risk of developing cancer. The company plans to test more wells, inspect the liner of the leaking fuel pool, and install additional monitoring wells.
>
> 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 : 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. 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 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 accidentally 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 serious accidents, including a core damage events. Therefore, to properly maintain the aging Indian Point 2 & 3 facility any and all compromised pipes must be replaced, including, but not limited to, the pipes under the reactor where information from discussions with Indian Point workers leads Stakeholders to believe seals may be leaking.

The insufficiency of a reliable aging management program in the LRA for Indian Point 2 & 3 increases the exposure risks to plant workers and the public during the 20 year period of new superseding license. It further greatly increases the potential for a significant nuclear incident at the Indian Point facility.

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. This presents a manifestly significant and increased risk to public health and safety.

The NRC itself has expressed concerns on this very issue relative 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 determine the potential severity of the effects of reactor water coolant environment on fatigue. Further, where conditions warrant, the NRC directed license renewal applicants to provide a proper aging management plan to deal with said fatigue issue. This concern is included in discussions found in NUREG/CR-6674.

The LRA for Indian Point 2 & 3 makes a brief reference to reliance on a nuclear regulator approach to this significant issue, yet fails to identify with specificity an aging management plan which deals with the unique site specific problems and 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.

Entergy's Indian Point facility (IP1, IP2 and IP3) has numerous serious leak issues. It is well established 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. 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 and IP3, 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, Exhibit GG1)

The poorly defined and inadequate aging management plan for Indian Point 2 & 3, as it 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.

Conclusion

Stakeholders contend that the LRA for Indian Point 2 and Indian Point 3 fails to include an adequate aging management plan as required by §10 CFR 54.21(a)(3) for piping.

Such piping is currently leaking radioactive effluent at 30 years of usage. Over the proposed time span of a 20 year superseding license, this becomes 60 year old piping.

The LRA accordingly does not provide an aging management plan for piping that assures adequate protection of 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."

CONTENTION #38: Microbial action potentially threatens all the stainless steel components, pipes, filters and valves at Indian Point (issue 99 of EIS)

In the Environmental Report section of the LRA Entergy's states that

there are no impacts from Microbiological (Thermophilic) Organisms [10 CFR 51.53(c)(3)(ii)(G)] However, eyewitness evidence suggests this representation is inaccurate

It has been reported by underwater divers who replaced the steel roller bearings on the traveling water screens for Indian Point that there were pit marks and holes in the stainless steel visible to the naked eye. The corrosion series 400 stainless steel roller bearings have been replaced with Inocel. However, this metal has also been compromised by the microbes in the river water. on the traveling water screens were caused by corrosive microbes and a lack of maintenance.

This same water is pumped though certain underground pipes in the plant, most of which have not been fully inspected. The recent April 7, 2007 leak of tritiated steam on the "clean" side of the plant may be a result of thinning and pitting of buried pipes due to microbial corrosion.

The microbial corrosion potentially effects certain the 400 series stainless steel, inspected and uninspected, components, pipes, filters, and valves at Indian Point. The ability of Applicant to maintain a safe, once through, or closed system that does not contaminate the environment is accordingly jeopardized.

Rapid, unmonitored, corrosion caused by microbes, can lead to a significant release of radioactive nuclides into the air, water, or ground. Corrosion that is clearly noticeable with the bare eye at a nuclear plant, as has occurred at Indian Point, presents a serious safety hazard and constitutes more than sufficient evidence that important systems and structures at Indian Point 2 & Indian Point 3 will be unacceptably vulnerable during the proposed 20 year new license period.

Stakeholders contend that the issue of microbial corrosion is a plant wide epidemic, and the LRA does not present a comprehensive aging management plan for the proposed 20 year superseding license period, therefore the NRC should deny the Applicant's LRA as being inadequate and incomplete.

CONTENTION #39: Indian Point 1 leaks constitute a violation of SafeStor and since components of IP1 are used in the operation of Indian Point 2, the LRA's failure to address these leaks and the interfacing IP 1-IP2 systems renders the LRA inaccurate, incomplete, and invalid

On December 1, 2005, Entergy reported to the NRC that an initial sample from a new monitoring well five feet from the wall of the Indian Point 2 Spent Fuel Pool showed tritium levels in the groundwater at thirty

times the EPA limit, the highest level of tritium contamination yet discovered.

The NRC announced that preliminary tests of tritiated water found in the Indian Point 1 (IP1) Pool Collection System contained too much tritium to be only from the IP1 Pool, alone, indicating that tritium-laced water is being collected in the IP1 Drain from another, unknown source.

Applicant then 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 of the SR-90 groundwater contamination. This flags the issue that Energy is also in violation of the term of their SafeStor Agreement of IP1.

When Entergy began trying to remediate the underground leaks by pumping the radioactive contamination out of the ground, it ended up causing even more radioactive material to be released. As a result, the NRC ordered Entergy to stop removing the radioactive effluent from ground, and to only monitor the radiation release. See exhibit JJ

Indian Point is located on the bank of the tidal Hudson Rive. Allowing the Radiation Leaks and the radioactive contamination to remain in the ground during the 20 year new superseding license period, means that the radioactive effluent will continue to be leached into the Hudson River.

This constitutes an untenable threat to, not only the environment, but to human health, 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 downriver, and the County of Rockland (which has a serious and growing serious water shortage problem) is currently planning use the Hudson River for drinking water. (Rockland has a serious and growing water shortage problem and has recently received a proposal from United Water for a desalination plant.

Doing nothing or dumping the radioactive effluent into the Hudson River as Entergy plans, are not acceptable alternatives. Letting radioactive water seep or be released incrementally into the Hudson River is not a proper form of mitigation, particularly for radioactive isotopes that remain toxic for decades, and in some cases centuries.

This proposed course of deliberate pollution, belies either an astonishing lack of lack of scientific knowledge on the part of both Entergy and the NRC, or a profound disregard for human health. In either case, Entergy has not presented any proof that it can properly manage Indian Point's radioactive pollution and waste products

Indian Point Unit 1 has been shut down for over three decades and is not cited under the LRA, however IP 1 is substantially affected by and

affects the operation of Unit 2. The dynamics of the interface between these Indian Point units creates an avalanche of mixed of safety, technical and environmental issues which introduce substantial additional complexity into the renewal proceedings.

By failing to address the IP1 leaks and by failing to include IP1 components and systems, the aging management plan is invalid and the LRA is incomplete, inaccurate and fatally flawed.

By disregarding of IP1 in the LRA, Applicant also defeats Stakeholders' rights of Intervention and Hearings, promulgated under the Federal Administrative Procedures Act.

CONTENTION 36 : <u>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.</u>"

Stakeholders contend that 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 <u>License Renewal Interim Staff Guidance</u> 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 disregarding the recommendation of the NRC, and even though the NRC incorporated it in next revision of License Renewal Interim Staff Guidance Supplement 1 to Regulatory Guide 4.2, "Preparation of Supplemental Environmental Reports for Applications to Renew Nuclear Power Plant Operating Licenses." The Applicant failed to address not just the rule, but also failed to address the trade guidance documents as well. (see Exhibit BB). The New York State Notice of Intention to Participate and Petition to Intervene and Supporting Declarations and Exhibits, filed on November 30, 2007 is referenced and incorporated as if fully set forth herein.

CONTENTION # 41 : Entergy's high level, long-term or permanent, nuclear waste dump on the bank of the Hudson River.

Issue Statement and Background:

The Indian Point site was never envisioned to be a nuclear waste dump. Stakeholders advance that the costs and impacts of indefinite, long term, and very possibly permanent, nuclear waste storage, were never conceived at the time Indian Point was first licensed and must be taken into consideration by the regulator now. Stakeholders further assert that failure

of the LRA to present a detailed plan for the management of all of the nuclear waste at Indian Point 2 & Indian Point 3 renders the LRA insufficient and is a ground for the denial of a new superseding license.

At the time the plans for Indian Point 1, 2 & 3 were being conceptualized, the assumption was made that, relatively soon after the nuclear plants commenced operation, scientists would come up with some safe and economically feasible way of disposing of nuclear waste. This optimistic assumption still prevailed when the original licenses for the plants were granted.

Unfortunately, after almost half a century of scientific effort, there remains no broadly accepted solution to the nuclear waste problem. As a result, the present plan of Entergy is to store the radioactive waste generated from Indian Point indefinitely on the site of Indian Point 2 and Indian Point 3. NRC staff has acceded that it is likely that high level radioactive waste (HLRW) and low level radioactive waste (LLRW) will need to be stored onsite at Indian Point for a period in excess of 100 years.

Thus, what was originally characterized as "interim" storage has turned in fact into "long term," and very possibly "permanent" radioactive waste storage.

The costs and impacts of indefinite nuclear waste storage at Indian Point will add an enormous and long-term burden to the state of New York, and to the Hudson Valley region, in particular, and must be fully evaluated in the Applicant's LRA EIS. The evaluation of this burden must also take into account the ongoing, unidentified and unremediated spent fuel leaks, the likelihood of future spent fuel leaks, and the proposed continued production of high level radioactive waste for an additional 20 years.

The spent fuel pools at Indian Point 2 & 3 are nearly full. At present, approximately over 2,000,000 pounds of high level radioactive waste is maintained on site. During the 20 years of the proposed new superseding license, that volume will increase by approximately 1,000,000 pounds, due to the up-rate of both plants. Applicant is planning to supplement the spent fuel pools by constructing an Independent Spent Fuel Storage Installation (ISFIS) Facility, a nearly 20,000 square foot concrete storage pad to hold 78 Holtec International HI-STORM 100S(B) dry casks. This constitutes a new use of the site.

Stakeholders contend that, since it is now apparent that there is no reliable plan for the off site disposal of Indian Point's nuclear waste, the LRA must define an aging management plan for such waste and the site

must be evaluated and licensed for long term or permanent high level radioactive waste storage.

This is both an issue of law and fact that must be heard by the ASLB prior to approval of the proposed new 20 year license.

A, The Environmental Costs and Impacts associated with the storage of both HLRW and LLRW waste streams, and of bringing onsite storage facilities into compliance with NRC rules and regulations promulgated for LLRW storage facilities, such as Envirocare in Utah, must be included within the scope of the EIS for Entergy's LRA for Indian Point 2 &3.

The structural integrity of the spent fuel pools are an issue of fact that must be fully evaluated, because their purpose to prevent radioactive effluent from being released unmonitored into the environment has been compromised. Spent Fuel leaks have plagued the plant since September 20, 2005, Analysis of soil samples taken in the vicinity of the cracks in the Indian Point 2 pool wall, in September 2005, indicated high levels of cobalt 60, cesium 134, and cesium 137, isotope activity which indicates the spent fuel pool water was leaking.

The following are summaries from the Journal News, the paper of record, of leak events since September, 2005:

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 assures the public there is no "immediate risk to public health or the environment." 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 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.

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. Exposure to tritium increases the risk of developing cancer. The company plans to test more wells, inspect the liner of the leaking fuel pool, and install additional monitoring wells.

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 : 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.

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 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.

April 24, 2007 A new leak of the radioactive isotope tritium has been 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 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.

September 29, 2007 Federal nuclear regulators have ruled that Indian Point didn't properly store a small amount of uranium 235 or document its movements stringently enough - although both violations are considered to have "very low safety significance." The weapons-grade uranium, located in 32 detectors used to monitor nuclear reactors, was stored in a spent fuel pool without being properly sealed from potential tampering.

Although Entergy continues to search for the source of the leaks, it has publicly maintained that all the leaks are coming from the spent fuel pools, However well test results showed that the "marker" isotopes found in some of the wells are not characteristic of the spent fuel pool water. This indicates that the leaks are not only from the spent fuel pools, but from other "hot" parts of the plant, such as the reactor or the reactor cooling system.

Well Tritium	Sample Date	Location
MW-34 63,900 p/CI/l	12/13/2005	Transformer Yard
MW-35 42, 300 p/CI/l	12/13/2005	Transformer Yard
MW-33 142,000 p/CI/l	12/13/2005	Transformer Yard

Additionally, Storm Water Drains Test results showed detectable levels for tritium in the storm drains ranging from less than 2000 pCi/L in sample locations 17, 18 and 19 to a high of between 12,000 and 51,000 pCi/L at sample location MHI-6 near monitoring well MW-11. The remaining wells tested between 2000 and 5300 pCi/L. This indicates that tritium is leaving the site. Storm drains at Indian Point flow into the discharge canal into the Hudson River. Any and all unmonitored releases place the Applicant in violation of NRC regulations governing releases of radioactive liquids from nuclear power plants into the water, ground or air. Code of Federal Regulations, Title 10, Part 50, Appendix A:

> ***Criterion 60--Control of releases of radioactive materials to the environment.*** **The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be**

provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 64--Monitoring radioactivity releases. **Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.**

§ 50.36a Technical specifications on effluents from nuclear power reactors.

(a) In order to keep releases of radioactive materials to unrestricted areas during normal conditions, including expected occurrences, as low as is reasonably achievable, each licensee of a nuclear power reactor will include technical specifications that, in addition to requiring compliance with applicable provisions of § 20.1301 of this chapter, require that:

(1) Operating procedures developed pursuant to § 50.34a(c) for the control of effluents be established and followed and that the radioactive waste system, pursuant to § 50.34a, be maintained and used. The licensee shall retain the operating procedures in effect as a record until the Commission terminates the license and shall retain each superseded revision of the procedures for 3 years from the date it was superseded.

(2) Each licensee shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases.
NRC's Regulations.

§ 20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that -

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003 NRC's Regulations.

§ 20.1302 Compliance with dose limits for individual members of the public.

(a) The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in § 20.1301.
NRC's Regulations.

Appendix B to Part 20--Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

The columns in Table 2 of this appendix captioned "Effluents," "Air," and "Water," are applicable to the assessment and control of dose to the public, particularly in the implementation of the provisions of § 20.1302. The concentration values given in Columns 1 and 2 of Table 2 are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem or 0.5 millisieverts).

Applicant has failed to comply with the above NRC regulations

that require *all* releases of radioactive material into the air and water be fully monitored. *Any* unmonitored release, no matter what the size, violates these regulations.

B. Stakeholders contend that the leaks from the spent fuel pools indicates that the structural integrity of the 30 year old concrete is compromised, and the LRA does not delineate an adequate aging management program to maintain these significant structures for the additional 20 year period of a new superseding license.

The concrete's "compromised structural integrity may inhibit a component from meeting its performance requirements (active or dormant cracks); e.g., diminished leak tightness or shielding, required to protect public health and safety.

The Hudson River water used for cooling at Indian Point and in the spent fuel pools is brackish, salty water, that has shown evidence of microbial biological elements that cause corrosion. This danger is recognized by the NRC:

> "Concrete exposed to a marine environment may deteriorate as a result of combined effects of chemical action of sea water constituents on cement hydration products, alkali-aggregate expansion, if reactive aggregates are present, crystallization pressure of salts within concrete, if one face of the structure is subject to wetting and others to drying conditions, frost action in cold climates, corrosion of

embedded steel reinforcement, and physical erosion due to wave action or floating objects". NUREG/CR-6927, ORNL/TM-2006-529 Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures- A Review of Pertinent Factors, February 2007 (p3)

"Cracking occurs in virtually all concrete structures and, because of concrete's inherently low tensile strength and lack of ductility, can never be totally eliminated. Cracks and crack patterns have different characteristics depending on the underlying cause. Cracks are significant from the standpoint that they can indicate major structural problems (active cracks); provide an important avenue for the ingress of hostile environments (active or dormant cracks); and may inhibit a component from meeting its performance requirements (active or dormant cracks) (e.g., diminished leak tightness or shielding". NUREG/CR-6927, ORNL/TM-2006-529 Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures- A Review of Pertinent Factors, February 2007 (p17)

Corrosion and cracking of aging concrete can be increased by salt crystallization, freeze/thaw, thermal exposure and thermal cycling, acid, alkali/silica, microbial/biological attack, vibration/fatigue, chemical interactions, irradiation. All these factors are present in the spent fuel pools at Indian Point. The briny salt water of the Hudson River is in the concrete spent fuel pools. "Although continuing the service of a nuclear power plant past the initial operating license period is not expected to be limited by the concrete structures, several incidences of age-related degradation have been reported." NUREG/CR-6927, ORNL/TM-2006-529 Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures- A Review of Pertinent Factors, February 2007.

The leaks from the spent fuels of Indian Point 1 and 2 indicate that there is serious age-related degradation. "Examples of some of these problems include corrosion of steel reinforcement in water intake structures, corrosion of post-tensioning tendon wires, leaching of tendon gallery concrete, low prestressing forces, and leakage of corrosion inhibitors from tendon sheaths. Other related problems include cracking and spalling of containment dome concrete due to freeze-thaw damage, low strengths of tendon wires, contamination of corrosion inhibitors by chlorides, and corrosion of concrete containment liners

Moreover, the NRC has recognized that, as nuclear plants age, the "incidences of degradation are expected to increase, primarily due to environmental effects". B.10 NUREG/CR-6927, ORNL/TM-2006-529 Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures- A Review of Pertinent Factors, February 2007 (p95)

Stakeholders contend that the structural integrity of the spent fuel pool and aging management of the same are squarely within scope and are matters of fact.

Stakeholders contend that the Applicant's LRA is incomplete, because it does not address the compromised structural integrity of the spent fuel pools, as evidenced by the ongoing leaks from unknown sources and

locations in the spent fuel pools. Applicant must set forth in detail an aging management plan to fully restore the structural integrity of the spent fuel pools, stop unmonitored releases, and remediate the leaks during the proposed new license period.

Additionally all issues, including costs and environmental impacts must be]clearly delineated in the EIS of the LRA prior to approval of the proposed new license.

If such complete inspection to identify the source of the leaks in the spent fuel pools, and complete restoration and remediation of the site is not feasible prior to approval of the proposed new license then the NRC must deny the Applicant's LRA.

CONTENTION # 42 Dry Cask Storage (Issue 83)
The Independent Spent Fuel Storage Installation (SFSI) being constructed at Indian Point for the purpose of holding the overflow of nuclear waste on site for decades, and probably more than a century, must be fully delineated and addressed in the aging management plan and, moreover constitutes an independent licensing issue.

The use of the Indian Point site for construction of the Independent Spent Fuel Storage Installation constitutes an entirely new and distinct use of the land, which was never contemplated when the site was first licensed. This new use will continue throughout the entire new license period, and very likely long thereafter.

Stakeholders contend that since this use of the land, on the bank of the Hudson River, for so-called "interim" spent fuel storage is not included in the Generic Environmental Impact Statement, a full and complete, site

specific EIS must be included in Applicant's LRA.

The dry cask storage pad design, and proposed configuration of the spent fuel casks, is considered Category 1 in 10 CFR 51 Appendix A subpart B. However, given the multiple emergent issues, it should be included in the SEIS as a Category 2 issue. This is an issue of fact which is affected by the following issues:

(1) there exists no current solution to the problem of long term permanent storage of nuclear waste;

(2) multiple spent fuel pool leaks and age related issues require design load changes to the ISFSI.

(3) the Barnwell LLWR storage facility will be closed to Indian Point;

(4)the possibility of unanticipated fissures in the pad;

(5) potential mixing of fuels from different units in loading of the casks and in the casks themselves, including waste from IP 1;

(6) NRC and industry research (as well as the reality that – even if Yucca opens – it cannot accept 60 years of Indian Point waste), pointing to the likelihood that Indian Point's radioactive waste will be sitting on site for many decades, and probably in excess of 100 years; and

(7) the need for a full and complete soil analysis which takes into

consideration the extent of contaminated soil requiring remediation, possible changes to the soil resulting from storm surge, and other effects from global warming, and new seismology studies, all of which are relevant the stability of the dry cask pad and casks.

The ISFSI,, also known as dry cask storage, estimates capacity of 75 Holtec 100 High Holtec Storm Casks, 18 ft high x 14 ft in diameter, 2.5 feet apart. Each cask will hold 32 PDR fuel assemblies, with a total of 2,400 fuel assemblies.

The casks will not be bolted down to the pad, even though ISFSI is believed to be situated virtually on top of the Ramapo Fault line, and even though the Indian Point region is one of the most congested air traffic regions in the nation. Significantly, Holtec International, the dry cask manufacturer, is trying to persuade DOE and the NRC to use an alternative cask design called TAD that would allow nuclear waste awaiting final disposition to be stored, in the interim, in below-surface boreholes rather than on more conventional above-ground pads. Kristopher W. Cummings, a manager for Holtec is reported as saying the TAD system layout would better shield waste canisters from earthquakes and from airplane or missile

crashes. This is a *de facto* admission by Holtec that its casks are vulnerable to earthquakes, airplane crashes and terrorist attack.

The concrete pad on a .5 acre pad with a 100 meter buffer of controlled land, uses approximately 40 acres of 239 acres site in Buchanan. The casks are 3 ft thick made of carbon steel inside concrete, which is corrosive.

In the event an earthquake or plane crash rocked Indian Point, the unbolted Holtec casks sitting on the bank of the Hudson River could tip over and roll into the Hudson or crack open and release radioactivity, causing enormous environmental impacts.

It is estimated that each plant will require one cask per year, during the new superseding license period, assuming there is no need to change fuel assemblies because of power uprate or other problems. The capacity of spent fuel at the IP 2 pool is 1374 assemblies, and at the IP 3 pool is 1345 fuel assemblies. Both pools are filled to near-capacity (Exhibit SS).

The IP 1 spent fuel needs approximately 5 MPC (casks) and the IP 1 spent fuel pool must be emptied immediately due to the strontium leaking from it into the groundwater and the Hudson River.

The additional time added by a new superseding license would mean, that, from this point forward, Indian Point 2 it will be in operation for over

25 more years, and Indian Point 3 will be in operation for over 27 more years.

This will leaves space for only some 20 additional casks. But, the inability of Entergy to find the leaks at the IP2 spent fuel pool, means there is a very real possibility that over the upcoming decades either or both of the spent fuel pools may need to be emptied to locate and remediate the leaks. It requires approximately 43 casks to empty Indian Point 2, and approximately 42 casks to empty Indian Point 3, based on the conference with NRC staff and Richard Barkley. (Exhibit SS)

In the event the IP 2 and or IP 3 spent fuel pool, or both, must be emptied, due to structural damage as evidenced by the current leaks, the dry cask pad may not be large enough to adequately store the high level radioactive waste generated during the 20 year new superseding license period.

Further it must be considered that in the event an additional dry cask storage pad is needed, there may not be adequate area on the site after the required closed-cycle cooling systems are installed.

Notably, representatives of the Commission acknowledged at public meetings held on June 27, 2007 and on September 19, 2007 that the NRC has not, to their knowledge, even evaluated whether the ISFSI Facility at

Indian Point site can safely hold all of the spent fuel it may need to contain.

In addition, the possible need to place even more high level radioactive waste in Holtec casks that do not meet the current required seismic standards of 5.5, much less the updated standards that should be made, must be transparently and comprehensively reviewed in the EIS.

Although the issue of spent fuel storage is an industry wide problem, the limited space of the Indian Point site and the proximity to New York City require that full plant-specific EIS mitigation measures be comprehensively presented and reviewed.

The increased spent fuel and dry cask storage at Indian Point must be considered as new information with potentially substantial adverse effect on the environment and human health for not just the 20 year new superseding license period, but for generations to come. Accordingly, any aging management plan for the Indian Point 2 & 3 should have fully delineated and addressed the issue. The Applicant's LRA did not, and should therefore be denied.

Further, since the plan for "interim spent fuel storage" is actually a plan for indefinite, permanent, high level radioactive waste storage, this new use of the site during the new license period, must be considered as a

Category 2 issue and requires full plant-specific analysis for Indian Point in the EIS.

Stakeholders contend that the NRC should not approve a new superseding license for the Applicant to operate Indian Point 2 & 3 for an additional 20 years, as it will only increase the amount of unplanned high level radioactive waste storage on the banks of the Hudson River, thereby decreasing public health and safety.

CONTENTION 43: The closure of Barnwell will turn Indian Point into a low level radioactive waste storage facility, a reality the GEIS utterly fails to address, and a fact which warrants independent application with public comment and regulatory review.

Barnwell South Carolina ("Barnwell"), operated by EnergySolutions, is the only radioactive waste disposal site in the United States that is currently operating and accessible for all classes A, B, and C (and greater than C on a case-by-case basis) of low level radioactive waste generated by nuclear power plants.

EnergySolutions has recently announced that Barnwell will be closed for use by states other than New Jersey, Connecticut and South Carolina after June 2008, and will, thus, cease to accept LLRW waste from Indian Point. (Specifically, some less concentrated, Class A LLRW may still

go to the Class A-only waste site that EnergySolutions owns and runs in Clive, Utah, but Classes B and C will not have a place to go after June 2008. Currently the DOE is doing an EIS on the disposal of greater than Class C waste.)

Thus Indian Point will also, *de facto*, be turned into a low level radioactive waste storage facility. This too is an issue the GEIS completely fails to address.

The low level nuclear waste disposal capacity at Indian Point is relatively small and Indian Point is not currently permitted to increase the low level waste disposal capacity.

Most critically, the transformation of the site into a repository for Indian Point 2 & 3 LLRW, will have significant environmental impacts and long term health effects on the population of the surrounding communities for generations.

This reality has been understood by the Commission. To wit, in a letter from the NRC to all licenses dated August 1, 1985, U.S. Nuclear Regulatory Commission Commercial Storage at Power Plant Sites of Radioactive waste Not Generated by the Utility HPPOS-092 PDR9111210185, W.J. Dircks states that:

"NRC is opposed to any activity at a reactor site that is not

supportive of authorized activities. Interim storage of low-level radioactive waste matter of policy, NRC is opposed to any activity at a nuclear reactor site which may divert attention of licensee management from its primary task of safe operation or construction of the power reactor.

The operator must demonstrate that the increased use of the low level waste facility does not involve a safety or environmental question, and that safe operation of the reactor will not be affected. The licensee must consider:

1. Direct impacts of commercial storage activities on reactor operations during normal and accident conditions.

2. Diversion of utility management and personnel attention from safe reactor operation.

3. Combined effects of onsite and offsite dose during normal and accident conditions.

4. Influence on effectiveness of both reactor emergency plans and reactor security plans.

5. Financial liability provisions, including impact on indemnity coverage.

6. Environmental impact of the storage facility, including potential interaction with the generating station. In addition, the following issues must be considered:

a. Safety of the commercial storage operation.

b. Environmental impact of the storage operation in sufficient detail for NRC to establish the need for an Environmental Impact Statement.

c. Financial assurance to provide for commercial storage operation and decommissioning including any necessary repackaging, transportation and disposal of the waste.

Moreover, with specific regard to the known radioactive leaks and planned refurbishment of the reactor vessel heads, the EIS must include a comprehensive review of the disposal plan of the old, highly irradiated and contaminated reactor vessel heads. Once again this is new information, for which Entergy has failed to include refurbishment or a disposal plan in Environmental Supplement E. Therefore the Applicant's LRA is incomplete and inaccurate.

Conclusion: The closure of Barnwell will turn Indian Point into a low level radioactive waste storage facility, a reality the GEIS utterly fails to address, and a fact which warrants independent application with public comment and regulatory review

Given the apparent reality that Entergy will be transforming Indian Point into use as a Class B and C (and possibly class A) radioactive waste storage facility, a fully independent application and review of such a change must be commenced for public and regulatory comment and consideration, prior to approval by the NRC of a new 20 year license for Indian Point 2 and Indian Point 3. In the event such a low level waste disposal storage facility is not approved, then the Applicant cannot continue to operate the plant.

Stakeholders contend that the NRC cannot approve the LRA without the Applicant first obtaining a license to dispose of low level waste that will be produced during the 20 year new license period.

CONTENTION 44 : 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: Stakeholders asserts that the Applicant's decommissioning trust fund balances are inadequate and insufficient to properly decommission the site, as required by 10CFR 54.3and 10CFR 50.75 to restore the site, and are not addressed in the Applicant's LRA, including but not limited to, removal of underground radioactive contamination in the groundwater and bedrock under a large portion of the Indian Point site. Therefore, due to the inadequacy of the decommissioning trust funds, the NRC cannot approve a new superseding license for an additional 20 years.

Basis for Contention

The Indian Point 2 and Indian Point 3 decommissioning trust fund have not been adjusted to take into consideration the enormous, underground radioactive contamination accidentally discovered in 2005. The costs for decommissioning, either DECON, ENTOMB or SAFESTOR must be

adjusted to reflect a significant change in the contamination streams, including, but not limited to, increased site contamination during the proposed additional 20 year superceding license period, and the current large underground radioactive leaks, as required by:

> § 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 1, 2 & 3, provide indicia of continued and future leaks during the additional 20 year superceding license period. 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 for the large amounts of underground radioactive effluent must be included, and the projected increase radioactive contamination, as part of the aging management and updated decommissioning plan in the LRA. However, it is not.

Per well-established law, where there is new and significant information suggesting that the basis for the original Generic Environmental Impact Statement (GEIS) conclusion is no longer valid, and specifically related to Indian Point, that conclusion is invalid because of known large leaks caused by failures in structural integrity, greatly change the impact and

costs of decommissioning Such new information regarding additional costs has not been incorporated into the LRA.

Neither in Entergy's LRA Environmental Report(Appendix E), nor the Generic Environmental Impact Statement (GEIS), provide a reasoned analysis of, the new and significant information (leaks) and their impact on the original findings with regard to the Decommissioning trust funds.

The GEIS states that the impacts associated with onsite land use are small Category 1 issues The basis for this assessment is the assumption that the land used for storage of nuclear wastes at the generic reactor site will not exceed 30 years after the end of the license term and is based on a zero leak assumption. This is a flawed assumption, and it is invalidated by the reality that the Indian Point 1, 2 & 3 are already leaking unmonitored radioactive effluent into the bedrock, groundwater and Hudson River. By relying upon the misdirected assumption that the decommissioning of Indian Point will be generic, the LRA fails to take into account the current leaks into the bedrock that will dramatically increase decommissioning costs at this site, thereby causing the impacts to be large.

a. 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. 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, or as recently proposed dumped into the Hudson River.

At a recent annual assessment NRC meeting in Croton, NY, NRC officials stated that since the radioactive leaks are in the bedrock under the plant, and since it is impossible to dig the radioactive contamination out, and infeasible to blast it out, therefore in order to decommission the site the tritium, cesium and strontium will have to be chiseled out 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 certainly require additional expense and protective actions as well as permission from OSHA. Such remediation work would be required 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.

b. 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 (Exhibit Y).

c. 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 and adds additional cost to properly decommission the plant. Additionally, the NRC has been discussing plans to store both Low Level Radioactive Waste (LLRW) and High Level Radioactive Water (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 confirmed lack of protections associated with forced onsite storage of radioactive waste streams must be addressed in the license renewal process, if left uncheck the Applicant and NRC will continue to use the Indian Point site as a radioactive waste dump for both LLRW and HLRW. The costs of decommissioning have dramatically increased since the 1970s when the first plants were decommissioned. LLW disposal is one of the most expensive factors in plant decommissioning. There has been a 2000% increase in the cost of disposal of LLW from $13/ft3 in 1983 to $300/ft3 today at the Barnwell facility for example Costs range from $200/ft3 to $500 /ft, these costs will be increasing during the 20 year new license period because Barnwell will stopped receiving waste from Indian Point in 2008.

d. The spent fuel pools were not designed to meet m the basic minimum requirements for structural stability and integrity as is outlined in the citing criteria for new reactors. The concrete and steel rebar have become compromised and corroded. Therefore it becomes imperative that the structural degradation indicated by the leaks of both Spent Fuel Pools 1 and 2 and the underground piping of Indian Point 1,2, and 3, be addressed and remediated before the license renewal application is even considered. In the event complete detection of all the leaks and full remediation is not possible, the NRC cannot approve the Applicant's LRA for an additional 20 years of operation.

e. 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 should not 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 trust fund, prior to any approvals or issuance of a new 20 year license to the Applicant.

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 Stakeholders have raised a material matter of fact or law, thus meeting the burden for further review. The adequacy of decommissioning trust funds are not addressed in the ER or GEIS of the LRA, and fail to meet the requirements of the NRC 10CFR50.75. These are a material issues of fact and law, and a full hearing on decommissioning costs and the adequacy of decommissioning trust funds must occur prior to the NRC approving a new superseding license for 20 years for IP2 and IP3.

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 based on the NRC own decommissioning reports, that is adequate to establish that said contention is entitled to a further and complete review of the issues contained herein.

The Stakeholders assert that the NRC must deny Indian Point 2 and Indian Point 3's LRA because it does not address the adequacy of the decommissioning trust funds, or methodology of decommissioning. 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, directly affects the proposed superseding license within section 50.13 "completeness and accuracy of the information", as it affects the continued aging management and safe operations of Indian Point 2 and Indian Point 3.

Stakeholder contend that NRC's Regulations, including Title 10, Part 50, Appendix A 2 include the following that presently in non-compliance.

The actual design criteria the plant is legally required to comply with was found to be in of itself in apparent non-compliance. (CONTENTIONS #22 -26 are referenced and incorporated fully as if set forth herein).

Some of the new circumstances appear to be in non-compliance are:

> Criterion 60--Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 64--Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

§ 50.36a Technical specifications on effluents from nuclear power reactors.

(a) In order to keep releases of radioactive materials to unrestricted areas during normal conditions, including expected occurrences, as low as is reasonably achievable, each licensee of a nuclear power reactor will include technical specifications that, in addition to requiring compliance with applicable provisions of § 20.1301 of this chapter, require that:

(1) Operating procedures developed pursuant to § 50.34a(c) for the control of effluents be established and followed and that the radioactive waste system, pursuant to § 50.34a, be maintained and used. The licensee shall retain the operating procedures in effect as a record until the Commission terminates the license and shall retain each superseded revision of the procedures for 3 years from the date it was superseded.

(2) Each licensee shall submit a report to the Commission annually that specifies the quantity of each of the principal radionuclides released to unrestricted areas in liquid and in gaseous effluents during the previous 12 months, including any other information as may be required by the Commission to estimate maximum potential annual radiation doses to the public resulting from effluent releases.

§ 20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that — (1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under §

35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003 NRC's Regulations.

§ 20.1302 Compliance with dose limits for individual members of the public.

(a) The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in § 20.1301.NRC's Regulations.

Appendix B to Part 20--Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

The columns in Table 2 of this appendix captioned "Effluents," "Air," and "Water," are applicable to the assessment and control of dose to the public, particularly in the implementation of the provisions of § 20.1302. The concentration values given in Columns 1 and 2 of Table 2 are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem or 0.5 millisieverts).

To comply with NRC's regulations on doses to the public, one must monitor all releases of radioactive material to the air and water. The Applicant has failed to comply with the above NRC's regulations that require must <u>all</u> releases of radioactive material to the air and water be fully monitored. <u>Any</u> unmonitored release, no matter its size, violates the regulations .these NRC's Regulations.

Any unmonitored releases, no matter of size, violates the regulations. and places Indian Point 2 and Indian Point 3 in non-compliance with NRC regulations, therefore Stakeholders contend that the Applicant must first address and remediate the present non-compliant issues, set forth above, and the environmental issues associated with each, before the NRC can approved the Applicant's LRA.

CONTENTION 45 <u>Non-Compliance with NYS DEC Law – Closed Cycle Cooling "Best Technology Available" Surface Water Quality, Hydrology and Use (for all plants)</u>

State Permits and Licenses from State agencies, specifically DEC SPDES permits, are required to discharge thermal pollution into the state owned discharge channel, and required fish return pipe lines. See Exhibit 1. Easements from New York State are required for the issuance of a new superceding license for a 20 year period. These required permits must be included in the LRA, as they directly relate to the Environmental Costs of thermal pollution and to potable water quality as required by State law.

Indian Point 2 and Indian Point 3 are equipped with separate cooling water systems that withdraw water from the Hudson River and discharge that water back to the River through a shared discharge canal (a "once-through" cooling system). The water is taken into the cooling system,

circulates past the condenser coils to absorb waste heat from operation of the generation equipment, and is discharged back to the River at a higher temperature than at the intake. The Stations withdraw up to 2.5 billion gallons of water per day from the Hudson River, through three intake structures on the shoreline. The heated non-contact cooling water is discharged to the River through sub-surface diffuser ports located along the seaward wall of the discharge canal, south of the intake structures.

The cumulative effects of all discharges from IP2 LLC and IP3 LLC must be weighed, and their Environmental Impacts and Costs considered in the EIS Scoping process. It is impossible to understand the Environmental Impacts and Costs associated with Indian Point Discharges without looking at the whole, as well as its singular year effluents totals. The EIS Supplement states:

> Water use conflicts (plants with cooling ponds or cooling towers using make-up water from a small river with low flow) [10 CFR 51.53(c)(3)(ii)(A)]
>
> IP2 and IP3 are equipped with once-through cooling systems that utilize make-up water from an estuary on the Hudson River. IP2 and IP3 do not have or use cooling ponds or cooling towers. Consideration of mitigation is not required.

The way Entergy presents Surface Water Quality, Hydrology and Use

(for all plants) would seem upon its face to close this environmental issue. However the Applicant has misrepresented the issue by omission. In the original Environmental Impact Study for IP2 LLC and IP3 LLC, both plants made a COMMITMENT to go to a closed cooling system.

The SPDES permits for Indian Point 2 and 3 pursuant to ECL§17-0811, requires that State Pollution Discharge Elimination System ("SPDES") permits incorporate current limitation and standards to ensure compliance with water quality standards, accordingly, EPA required that facilities be retrofitted from once-through cooling systems, to closed-cycle cooling systems employing "best technology available".

Despite appeals by Entergy, New York State courts have upheld a decision requiring Indian Point to install "best technology available" prior to approval of a required discharge permit. New York State Department of Environmental Conservation (DEC) is preparing a final order for closed-cycle cooling system., Entergy's comments claiming that not mitigation is needed regarding once-through cooling is at best, misleading.

The population that is affected by this omission is the People of the State of New York, as they are the true owners and users of the Hudson River, which is affected by Indian Point the thermal pollution of a billion gallons of water a day, in violation of the Clean Water Act.

The thermal pollution significantly and adversely affects the larvae and fish populations of the Hudson River. Entergy not using the •best available technology with regard to closed cycle cooling is an issue of LARGE significance. Indian Point takes in a billion gallons of Hudson River water a day and super heats it 15 to 25 degrees before discharging it back into the Hudson, dramatically affecting the flora and fauna of the river.

Entergy also fails to present a complete analysis of compliance, and falsely submits in the ER on page 9-2 that

> Compliance with the SPDES Permits over previous years has been excellent. For example, there has never even been an exceedance relative to thermal discharge limits as identified in the Station's SPDES permit".

This is a misrepresentation, since Indian Point has been in non-compliance with the compliance status to the Environmental Protection Agency, Clean Water Act, and NYS DEC requirements to use the •best technology available , to prevent thermal pollution since it was enacted in 1975.

Therefore Stakeholders contend that until the Applicant installs closed-cycle cooling systems for Indian Point 2 and 3 and comes into compliance with the EPA – Clean Water Act and New York State DEC requirement a new license for 20 years cannot be approved.

CONTENTION 46: Omitted

CONTENTION 47: The Environmental Report Fails to Consider the Higher than Average Cancer Rates and Other Health Impacts in Four Counties Surrounding Indian Point.

The Stakeholders contend that the continued operation of Indian Point for 20 years during the new superceding license period, raises risks serious life threatening illness caused by increased radioactive exposure which raises a specific issue of law and fact.

During the proposed 20 year new superceding license period the reactor cores would maintain high levels of radioactivity in the core and approximately double the 3 million pounds of high level radioactive waste already at the site, worsening the consequences of a large-scale release after a mechanical failure or an act of sabotage. Many thousands would be stricken with acute radiation poisoning, cancer, genetic defects and serious health effects. Currently the amount of radioactivity that can be released from at the plant is equivalent to up to 70 times as much as Chernobyl , which resulted in over 1,000 square mile dead-zone, deemed uninhabitable for centuries. and a hundred of times as much as was released at Hiroshima, (Exhibit MM).

A national solution to the radioactive waste produced every day at Indian Point does not exist. Moreover, Barnwell, the only low level radioactive waste dump on the East Coast, will not be accepting waste from Indian Point. This means that during the proposed 20 year new superceding license period, the accumulation of radioactive waste at Indian Point will more than double, and greatly increase significant health issues. The License Renewal Application (LRA) does not address this significant change, nor does it address the health consequences.

The potential health and environmental consequences of 20 additional years of dumping radionuclides into the Hudson River, both monitored and unmonitored, combined with an analysis of the synergistic interaction of such radionuclides with other known Hudson River pollutants like PCBs, endocrine disruptors (including dioxins) and mercury; additional releases of radiation and other chemical toxins released by Indian Point into the environment (especially toxic metals like cadmium) upon populations most susceptible to radiation and toxic chemicals, such as women, adolescents, children, babies, breast-fed infants and the embryo/fetus are significant issues of fact.

During the proposed 20 year new superceding license period, Indian Point is more vulnerable to a meltdown from mechanical failure than most

reactors because of its age, and more vulnerable to a terrorist attack because of its proximity to New York City. Since the terrorist attack on the World Trade Center of September 11, 2001, much attention has been paid to the possibility of Indian Point as a potential target for attack. Recent government studies on terrorism has shown that worldwide terrorism is on the rise and is projected to continue to rise during the next 20 years.

The potential for a meltdown, or spent fuel fire, while not highly likely, is a reality. A recent report by Greenpeace entitled "An American Chernobyl" identified 200 near-miss accidents at American reactors in the past two decades.

Near Miss Accidents and Unplanned Shutdowns At Indian Point Since 2000

Date	Reactor	Description
February 15, 2000	Indian Point 2	Steam generator tube rupture
July 19, 2002	Indian Point 2	Degraded control room fire barrier
August 14, 2003	Indian Point 2 &3	Loss of offsite power, emergency generator failure and emergency communication failure due to NE blackout
September 4, 2004	Indian Point 2	Valve failure, improper water levels
September 15,2004	Indian Point 2	Valve failure
September 24, 2004	Indian Point 2	Valve failure
December 3, 3004	Indian Point 2	Welding problems
February 10, 2005	Indian Point 3	Control rods fail to load properly
May 18, 2005	Indian Point 2	Nitrogen gas in safety injection pump
October 7, 2005	Indian Point 3	Control rod malfunction
December 24, 2005	Indian Point 2	Faulty steam generator valve seal
July 5, 2006	Indian Point 3	Electrical relay main generator tripped
July 8, 2006	Indian Point 3	Electrical short under transformer
July 22, 2006	Indian Point 3	Electrical mishap of high-voltage wiring
August 24, 2006	Indian Point 2	Faulty valves cause drainage problem
November 16, 2006	Indian Point 2	Electrical conductor malfunctioned
February 8, 2007	Indian Point 2 &3	Intake water structure clogged

March 1, 2007	Indian Point 2	Steam generator feed pump malfunction
April 4, 2007	Indian Point 2	Steam generator malfunctions
May 30, 2007	Indian Point 2	Steam generator broken water valve

Estimates of Casualties. If a meltdown, or radioactive fire, that caused large scale releases of radioactivity occurred, vast numbers would suffer from acute radiation poisoning (in the short term) and cancer (in the long term). Several estimates have been made to calculate just how many would be harmed. In 1982, the Sandia National Laboratories submitted estimates to Congress for each U.S. nuclear plant in the case of core meltdown.

Estimated Deaths/Cases of Acute Radiation Poisoning and Cancer Deaths Near Indian Point, Following a Core Meltdown

Type of Effect	Indian Point 2	Indian Point 3
Deaths, Acute Radiation Poisoning	46,000	50,000
Cases, Acute Radiation Poisoning	141,000	167,000
Cancer Deaths	13,000	14,000

Note: Acute radiation poisoning cases and deaths calculated for a radius of 17.5 miles from the plant, cancer deaths calculated for radius 50 miles from the plant. Source: Sandia National Laboratories, Calculation of Reactor Accident Consequences (CRAC-2) for U.S. Nuclear Power Plants. Prepared for U.S. Congress, Subcommittee on Oversight and Investigations, Committee on Interior and Insular Affairs. November 1, 1982. Published in New York Times and Washington Post the following day.

The Sandia figures are known as CRAC-2 (for Calculation of Reactor Accident Consequences). CRAC-2 estimated casualties for Indian Point are one of the highest of any U.S. nuclear plant. The same calculus using

updated census figures would be far larger. Since 1970, the population within the 17.5 mile "peak-fatality" zone has grown significantly. Moreover, people well beyond a 17.5 mile radius will also suffer adverse health consequences.

More recently, the Union of Concerned Scientists prepared an estimate of casualties after a core meltdown from a terrorist attack. The 2004 report entitled "Chernobyl on the Hudson" estimated much higher casualties than did the 1982 Sandia effort. The Union's Dr. Edwin Lyman calculated that as many as 44,000 near term deaths from acute radiation syndrome within 50 miles and 518,000 long term deaths from cancer within 60 miles could occur, depending on weather conditions. (Source: Lyman ES, Chernobyl on the Hudson?: The Health and Economic Impacts of a Terrorist Attack on the Indian Point Nuclear Plant." Washington DC: Union of Concerned Scientists, 2004. www.ucsusa.org).

Due to the increasing population within the 50 mile radius of Indian Point during the 20 year new superceding license period, that number will continue to grow. The growth rates from 1970 to 2002 in surrounding counties are respectively Westchester 15.88%, Rockland 26.7% , Orange 72.11%, and Putnam 68%..

Stakeholders contend that Indian Point reactors routinely release radioactivity and persons living near Indian Point area exposed to these radioactive chemicals. Historically, Indian Point has a checkered record of contaminating the local environment.

Environmental Releases from Indian Point. All nuclear reactors routinely emit radioactivity into the environment in order to operate. These emissions take several forms. Accidental releases due to leaking equipment, which can include the cladding and welds of fuel rods in the reactor core, cracks and breaks in fuel that damages cladding, corroding pipes, and cracked steam generator tubes. These scenarios result in radioactivity released into the air and water. Radioactivity is also deliberately released into local water about every 18 months when reactors refuel. In fact, Indian Point will be releasing large amounts of radioactive waste into the Hudson River in the near future as they attempt to clean up the highly contaminated site of Indian Point 1.

Each utility is required by federal law to measure and report radioactive environmental emissions from nuclear reactors annually. From 1970-1993, the federal government produced a comparative listing of annual emissions for each U.S. reactor (it has since been discontinued). One measure of environmental emissions is known as airborne "Iodine-131 and

Effluents" or chemicals with a half life of at least eight days (and thus, are more likely to enter the body through breathing and the food chain).

U.S. Nuclear Plants with Highest Emissions
Of Airborne Radioactivity, 1970-1993

Plant	Location	Reactors	Emissions*
1. Dresden	Morris IL	3	97.22
2. Oyster Creek	Forked River NJ	1	77.05
3. Millstone	Waterford CT	2	32.80
4. Quad Cities	Cordova IL	2	26.95
5. Indian Point	**Buchanan NY**	**3**	**17.50**
6. Nine Mile Point	Scriba NY	2	14.67
7. Brunswick	Southport NC	2	14.50
8. Three Mile Island	Londonderry PA	2	14.43
9. Monticello	Monticello MN	1	12.48
10. Pilgrim	Plymouth MA	1	6.71

* Emissions expressed as curies of Iodine-131 and effluents
Source: Tichler J et al. Radioactive Materials Released from Nuclear Power Plants, annual reports. Upton NY: Brookhaven National Laboratory, NUREG/CR-2907.

The Indian Point total of 17.50 curies is the 5th highest of 72 U.S. plants. The total is greater than the 14.43 curies from the Three Mile Island plant in Pennsylvania, most of which was reported after the 1979 accident. Most of the Indian Point total occurred in 1985 and 1986, with a total of 14.03 curies from Indian Point 2. Several years later, the totals were changed to 1.90 curies; an inquiry to the U.S. Nuclear Regulatory Commission attributed the change to a "clerical error." While the original

figures are used here, using revised figures would still rank Indian Point as the 12th highest in the nation.

More recent data on emissions is now posted on the Internet by the federal government. Data for all U.S. reactors are listed from 2001-2004, by quarter, and by type of emission. Unfortunately, no information for Indian Point 2 is given, and data for Indian Point 3 is missing for various quarters.

But examination of types of airborne and liquid radioactive emissions with data reported for each quarter from 2001-2004 from Indian Point 3 is helpful in understanding the large variations over time. For example:

- Releases of fission gases from Indian Point 3 rose about six-fold from the fourth quarter 2001 to the first quarter 2002 (about 15-fold for Xenon-133, a type of fission gas), and about 100 times higher than a year earlier.

- Second quarter 2004 releases of airborne fission gases were much higher than typical quarterly 2003 releases

- The quarters with the highest liquid releases of fission and activation products were not necessarily those with the highest liquid releases of tritium

In addition to the monitored releases, Indian Point has released large amounts of unmonitored radioactivity into the environment, specifically into the groundwater. Since Indian Point 1 was closed in 1974, it was known that approximately 10 gallons a day of radioactive effluent was leaking from the plant into the groundwater, however when new

investigations into the leakage were commenced in 2005 after a leak was accidentally found at Indian Point 2 spent fuel pool, the estimated leakage from Indian 1 has been increased to approximately 30 gallons a day. Using these calculations, since 1974 approximately 150,000 gallons of radioactive effluent waste has been released into the environment, groundwater and ultimately into the Hudson River water and air. For years Applicant has failed to prevent unplanned releases, and has been unable to ascertain the source, extend and causes of radioactive leaks. The License Renewal Application utterly fails to adequately address either how the Applicant will effectively prevent increases in the annual unplanned releases from the plant, or how it will effectively prevent continued unmonitored releases during the proposed 20 year new superceding license period.

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."

Environmental Radioactivity Levels near Indian Point. New York State Department of Health maintains a water monitoring site on the Hudson River at Verplanck, which is just one mile south of the Indian Point plant. It

also measures radioactivity levels in water in Albany, on the roof of the Health Department building, as a "control", meaning the site is far from any nuclear plant. Average weekly levels of all alpha and beta emitters have traditionally been about 10 to 11 times higher in Verplanck than in Albany. It is virtually certain that this difference is due to the operations of Indian Point, as many of these alpha- and beta-emitting chemicals can only be produced in nuclear reactors. Source: New York State Department of Health, Bureau of Radiation Protection. Environmental Radiation in New York State, annual volumes.

Radioactivity Levels in Bodies near Indian Point. The question of how much man-made radioactivity enters human bodies was first considered in the 1950s, when the U.S. government sponsored studies that measured bone and teeth samples for Strontium-90, one of the 100-plus chemicals found in nuclear weapon explosions and nuclear reactor operations. A landmark study of baby teeth in St. Louis found that the average Sr-90 level for children born in 1964 (just as atomic bomb testing was stopped) was about 50 times greater than for children born in 1950. Furthermore, Sr-90 studies found that average concentrations in bodies plunged by about half from 1964 to 1969, after large-scale weapons testing in the atmosphere was banned. Similar studies of Sr-90 in bone and teeth in Europe found similar patterns.

(Sources: Rosenthal HR. Accumulation of environmental strontium-90 in teeth of children. In: Proceedings of the Ninth Annual Hanford Biology Symposium, Richland WA, May 5-8, 1969. Washington DC: U.S. Atomic Energy Commission, 1969. Health and Safety Laboratory, U.S.Atomic Energy Commission. Strontium-90 in Human Vertebrae. In: Radiation Data and Reports, monthly volumes, 1964-1969).

Over 500 teeth were collected and tested from the New York metropolitan area. Over half were from the four counties closest to Indian Point – Westchester, Rockland, Orange, and Putnam. The average local Sr-90 level was the highest in the area, and the highest of near six U.S. nuclear plants. Average Sr-90 decreased with distance from the plant, i.e., New York City was lower than the local area, and Long Island was lower than New York City.

Increases in average Sr-90 in baby teeth over the past decade were also highest near Indian Point. Children born in the late 1990s in the four-county area had a 38% greater average Sr-90 level than those born in the late 1980s. This increase indicates that during the proposed 20 year new superceding license period, the Sr-90 levels in baby teeth with continue to increase, first because the plants have been uprated and therefore produce more radioactive waste, and secondly because as the plant continues to age it will leak more radioactive waste into the environment, and most importantly health effects of radioactive exposure are CUMULATIVE.

This record of contamination raises health concerns, which are heightened when considering that, since 2000, in the four counties closest to Indian Point, cancer incidence is dramatically higher than average United States levels. The New York State Department of Health has made cancer incidence data available on the Internet for small areas, including counties and zip codes. The most recent data covers the period 2000-2004. The cancer incidence rate in each county exceeds the U.S. and state rates, for both genders (greater for females). In nearly all cases, the excess was statistically significant. If the rate for each county had been equal to the national and state rates, 3631 and 2090 fewer cancer cases, respectively, would have been diagnosed during 2000-2004. (Exhibit UU).

Childhood cancer incidence is 22% above the U.S. rate

Thyroid cancer incidence is 70% above the U.S. rate

Cancer incidence in the six towns within five miles of Indian Point is 20% greater than the rest of Rockland and Westchester Counties. (Public Health Risks of Extending Licenses of the Indian Point, Exhibit TT).

Closure and decommissioning of Indian Point at the end of the current license period, for Indian Point 2 in 2013 and Indian Point 3 in 2015, will result in decreases in cancer mortality, as it did near the closed Rancho Seco plant in California. In the event the NRC does not approve Entergy's

LRA and the plants close in 2013 and 2015, respectively, 5,000 fewer cancer deaths would likely occur in the next 20 years in Westchester, Rockland, Orange, and Putnam Counties.

The issue of pubic health and safety during a 20 year new superceding license due to increased exposure to radioactive releases is within scope of the Proceeding, as the underlying mandate of the NRC is to adequately protect public health and safety. Radioactive exposure is CUMULATIVE, and therefore must be evaluated over the period of 60 years, rather than 40 years.

Long-term exposure is typically used to describe chronic small exposures over many years, such as the kind to which populations residing in the vicinity of nuclear reactors are exposed. The National Academy of Sciences BEIR VII report and a 2003 report issued by the European Commission on Radiation Risk (ECRR) have concluded that significant biologic effects can result from such exposure. The threat is particularly acute with respect to a radionuclide like strontium which remains intensely radioactive for three decades; thus every exposure is an additive to prior, ongoing, exposures.

This is also true for tritium, which – since it is a radioactive form of water – also easily bio-concentrates in the food chain. Accordingly, people

who eat fish from a river into which tritium has been leaked could also ingest tritium, which will remain in the ecosystem for over two centuries. Notably tritium is most dangerous when it becomes organically bound in molecules of food. It can then become incorporated into cell molecules, including DNA. In such case, tritium has a longer biological half-life (from 21 to 550 days) and can remain in the body for years.

The material issues raised are based on the expert studies of National Academies of Science (NAS) report, Health Risks from Exposure to Low Levels of Ionizing Radiation, known as BEIR VII (Biological Effects of Ionizing Radiation) to be a landmark study. It was released in June 2005 and updated in 2006.

The National Academy of Sciences BEIR VII report concludes that women are significantly more vulnerable to radiation than are men and that the cancer mortality risks for solid tumors are almost 50% greater for women (though for leukemia, the risk estimates are higher for men). The BEIR VII panel was also in accord with the European Commission on Radiation Risk, in determining that the risk differential for children – especially babies and very young children – is even more dramatic. For instance, the cancer risk for male infants up to age one is 3 - 4 times that for

males in the age range of 20 - 50 exposed to the same amount of radiation. Female babies and children are even more vulnerable than males.

It is also well established that radiation is most potent to the rapidly dividing cells of babies in utero. (Gamma rays can pass through the fetus. Alpha and beta particles can be transmitted via the placenta.) Radiation interferes to a high degree with cell proliferation and such rates exist throughout prenatal development. The central nervous system may be at especially high risk. Central nervous system development starts during the first weeks of embryonic development and continues through the early postnatal period. This system is accordingly quite vulnerable for a very long period. The constellation of effects from injury to the developing central nervous system includes: mental retardation, autism spectrum disorders, learning disabilities, and ADD. Tissues that are particularly susceptible if exposed during normal periods of rapid growth (i.e., prenatal, early childhood and puberty) are the brain, thyroid, bone and breast. Moreover, infants are vulnerable to the transference of isotopes like strontium-90 will pass to a newborn during breastfeeding.

Notably, a Radiation and Public Health Project study published in the February 2003 issue of the Archives of Environmental Health examined rates of cancer of children living near operating U.S. nuclear reactors. The

study found that cancer incidence for children under age 10 living within 30 miles of each of the 14 nuclear plants in the eastern U.S. exceeded the national average. Incidence was particularly elevated for leukemia.

Of the 14 power plant regions studied, the childhood cancer rates in Rockland and Westchester Counties near the Indian Point plant was 4th highest (17.4% above the U.S. average).

Notwithstanding their special vulnerability, women, children, babies and the embryo/fetus are not given corresponding consideration in the regulatory framework which governs nuclear power plant emissions. The current outmoded standards do not incorporate the medical knowledge that has been attained during the past 20 years and continues to be based on •Reference Man , which is defined as a young adult Caucasian male. (The term actually derives from the standards created to protect the young, white, male scientists working in nuclear labs during the early post-Manhattan Project era.) The regulatory paradigm is also narrowly oriented to cancer, thereby ignoring neurological injury, genetic damage, and the wide panoply of other extensively reported conditions that can result from exposure to ionizing radiation.

Compliance with outdated standards does not adequately protect the health of the public. Additionally since radioactive exposure is

CUMULATIVE, the health effects of Indian Point must be evaluated over the period of 60 years, rather than 40 years.

The undersigned Stakeholders contend that the NRC must deny the Applicant's LRA because it fails to address adequate the protection of public health and safety, specifically the effects on the most vulnerable members of the population, embryo/fetus', babies, children and women. Additionally, the undersigned Stakeholders contend that the NRC must deny the Applicant's LRA because it fails to address adequate the protection of public health and safety from CUMULATIVE radioactive exposure for 60 years, during the current license and additional proposed 20 year new superseding license period.

CONTENTION 48 : Environmental Justice - Corporate Welfare

The nuclear industry enjoys financial incentives, far beyond what is available to other more environmental friendly renewable energy sources.

Between 1947 and 1999, the nuclear industry was given more than $115 billion in direct taxpayer subsidies, compared to a mere $5.7 billion for wind and solar over the same period. The Energy Policy Act of 2005, was filled with nuclear industry largesse with an additional $3 billion dollar subsidy to the mature nuclear industry that already has received the lion's

share of federal energy funds over the past 50 years. Stakeholders contend that these ongoing subsidies to the nuclear industry have resulted in a violation of Fair Trade doctrine.

Ratepayers and taxpayers are the victims of this violation of fair trade. Specifically, New York State taxpayers, and the residents surrounding Indian Point, are footing the majority of the costs for Emergency Preparedness, and due to the short fall in decommissioning trust funds, will be burdened by the cost of site clean up. At Indian Point, Entergy is making a profit of nearly $2 million dollars a day but does not adequately cover the costs of its plant's security.

Yet, as an example of the gross inequity and the violation of Fair Trade standards, last year Entergy's Chairman received a salary of $27 million dollars, even though Entergy New Orleans filed for bankruptcy, and received a government bailout of almost $300 million, while at the same time ratepayers in New Orleans were smacked with greatly increased electrical bills.

The deregulation electricity market since the Indian Point was originally sited, is a significant change and must be fully addressed in the LRA. Due to the deregulation of the electricity market where free trade is a core tenet, a fair analysis of increased costs and exposures to the community

as it relates to Indian Point. Economic subsidies from tax dollars are going to support nuclear energy facilities, such as Indian Point must be included in the LRA economic analysis.

The claim that nuclear power is cheap energy should be fully explored including but not limited to, operational costs, the costs of research and development, and costs borne by taxpayers by way of subsidies and research paid for through DOE hand outs to EPRI, and universities such as MIT.

Entergy's large stake in unregulated wholesale markets for nuclear energy give it a big edge over traditional utilities. With profits for its nuclear operations growing much more quickly than for its regulated utilities, Entergy plans to spin off its six unregulated nuclear plants, including Indian Point, into a different company. In fact, Entergy has filed a license transfer application, after filing the license renewal applications, for both IP2 and IP3 to Entergy Nuclear Operations.

Indian Point is no longer a public utility, but instead a private corporation who sole purpose is to make a profit for its shareholder. This corporate goal, outweighs the responsibility to protect public health and safety.

This is an issue of both law and fact, with regard to Fair Trade and equal protection under the law are being violated. Stakeholders and Ratepayers are at a distinct disadvantage in advocating for public health and safety, when Entergy has the ability to use the financial benefits it receives from government subsidies and taxpayer dollars to finance a powerful legal staff and a major public propaganda campaign.

In order to mitigate this imbalance, the NRC would be warranted in requiring the Applicant to pay for the legal expenses of the community Stakeholders, and require a comprehensive study of the actual costs to taxpayers for the operation of Indian Point, including, but not limited to:

a. Annual Federal, State and local Subsidies and tax credits

b. State and local pilot tax deferments

c. Price Anderson Insurance Liability Limitation-specifically the costs to citizens should an accident occur, since the act makes it impossible for citizens to insure against the losses that would be incurred from a significant nuclear incident or terrorist attack at the facility.

d. Costs of emergency preparedness (at all levels of government).

e. Costs of security for all nuclear facilities that are absorbed or offset by all levels of government.

f. Federal and state funded research and development. This is to include all research for the ENTIRE fuel cycle.

g. Costs of mining, including clean up of contaminated sites involved in the nuclear fuel cycle, including specifically Paducah and Portsmouth Gaseous Diffusion Plants. Further, all pay outs to former nuclear workers for health related issues should be included in this figure.

h. Cost of processing, including transportation at all steps of the process, governmental paid expenses associated with construction of fuel processing facilities, such as the Gaseous Diffusion Plants, and the proposed GNEP reprocessing plant. These costs should also include environmental restoration and clean up costs associated with these processing facilities, such as Hanford and other locations.

i. Costs of plant construction (including loan guarantees, and siting grants.

j. Costs of transportation

k. Costs of radioactive waste storage (which should include the monthly surcharge being added to our bills to cover the expected costs of off site storage).

l. Costs of decommissioning and returning site to green field.

m. Cost of health effects, including deaths associated with the entire fuel cycle, including up through the long term storage of nuclear waste streams.

n. Costs of regulatory enforcement not covered by licensee fees. As example, the $980 million dollar budget this year for the NRC.

p. Additionally, the extent that there are dramatically unequal subsidies and total life-cycle costs between Nuclear Energy production and energy efficiency and renewable energy sources, such as geothermal, photovoltaic and wind, must be comprehensively considered. True sustainable and renewable safe, forms of energy that are widely viewed as the energy technologies of the •future, as well as efficiency technologies and demand side options, must be considered in the EIS, including the replacement energy study by NAS commissioned by Nita Lowey.

Stakeholders contend that the issue of corporate welfare during the proposed twenty year new license is within scope as it addressed the economic basis. The gross imbalance in taxpayer subsidies to Indian Point, over all other forms of energy in the region, is a violation of the edicts of Fair Trade and right of equal protection.

Therefore Stakeholders contend that the Applicant's LRA is incomplete and inaccurate and therefore the NRC cannot approve the LRA.

CONTENTION 49: Applicant's LRA fails to consider the effects of global warming and Applicant has failed to present a plan for how it will either analyze or manage such effects during 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, the rules under 10 CFR 54, "License Renewal," require that Applicant set forth an aging management plan for the additional 20 years of operation of Indian Point. Such a plan must, under any logical application of the federal regulations, take cognizance of the realities of climate conditions.

Global warming is now accepted by the broad scientific community as a virtual inevitability. Anticipated conditions include long and severe

droughts, the warming of rivers, and more frequent and severe storms. Such effects need to be taken into serious and detailed consideration in the aging management plan for Indian Point.

Obvious likely effects include: a drop in the water level of the Hudson River, upon which Indian Point is dependent for cooling; the rise in temperature of the Hudson River; drought and heat waves, which increase fire hazard risks; an increased frequency and level of storms, including storm surges and flooding which can facilitate the corrosion and disintegration of buried piping and other plant components and systems; alterations in the terrain, particularly the composition and solidity of the ground and soil which forms the foundation for plant structures and systems; destructive wind and lightning storms which can disrupt off-site and on-site electrical power in a wide variety of ways; and the clogging of critical piping and intake structures from debris.

Stakeholders assert that Indian Point 2, LLC and Indian Point 3, LLC's License Renewal Application (LRA) does not include any plan, much less an adequate plan, to analyze, monitor and manage the effects of climate change.

There is a Strong Scientific Consensus that a Warming World is a Real Phenomenon Posing Real Danger

In a series of increasingly grim assessments, leading international climate scientists, including the U.N.'s Intergovernmental Panel on Climate Change (IPCC), have concluded that global warming is unequivocal. In addition to the IPCC, over the past 5 years, the American Meteorological Society, the American Geophysical Union, the American Association for the Advancement of Science and the National Academy of Sciences have determined that there is clear evidence of global warming.

While the speed and extent of global warming is unknown (due, in large part, to uncertainty over whether nations will take the steps needed urgently and immediately to avert the more catastrophic outcomes), rising temperatures and shifting weather patterns have already occurred, and are expected to increasingly occur over the next several decades.

The Predicted Conditions of Climate Change are Highly Relevant to an Extended Operation of Indian Point and May Not Legitimately be Ignored

The effects of climate change have already been felt by nuclear power reactors around the globe. During the extreme heat wave that struck Europe in 2003, more than a dozen nuclear plants in France, Spain and Germany were forced to shut down or reduce power due because their water cooling sources were too hot or water levels were too low. In the U.S., in 2006,

nuclear plants in Minnesota, Illinois and Pennsylvania were forced to slowdown and a plant in Michigan was forced to shutdown because of hot conditions. In August 2007, excessively warm river temperatures forced nuclear plants in Alabama and in Vermont (Entergy's Vermont Yankee) to cut power production.

(While environmental damage and violation of environmental laws have, so far, been the primary considerations for such power cuts, the danger presented to plant safety, which is reliant on adequate cooling, is also apparent. Moreover, the need for nuclear reactors to cease or reduce electrical output during heat waves, when the need of seniors and people with respiratory and other medical conditions for air conditioning is paramount, presents a genuine danger to public health and is a compelling reason for the NRC to acknowledge the benefits of other forms of power, especially solar power, as an alternative to Indian Point.)

There is no basis to assume that the Northeastern U.S., or the Hudson River Valley in which Indian Point is located would be immune to the effects of climate change. Indeed a July 2007 report issued by the Northeast Climate Impacts Assessment, a 50-expert panel, led by Peter Frumhoff of the Union of Concerned Scientists, determined that if heat-trapping emissions are not significantly curtailed, global warming will substantially

change critical aspects of the Northeast's climate. Summers in the region could warm by up to 14°F above historic levels by late this century. The report, *Confronting Climate Change in the U.S. Northeast: Science, Impacts, and Solutions*, (July 2007) by the Northeast Climate Impacts Assessment (NECIA) – a collaboration between the Union of Concerned Scientists (UCS) and a team of more than 50 independent scientists and economists – broadly looks at how the climate of New York and the 8 other Northeast states will be likely to change under low and high-emission scenarios.

Water levels in Lake Ontario, and Lake Superior are below long-term averages, and during a major drought, the water levels of the Hudson may reasonably be expected to fall critically low, especially during summer months. The warming of the Hudson River over the next several decades must also be factored into any calculus. Indeed, evidence suggests that the warming of the river has already begun. A study by researchers at the Institute of Ecosystem Studies found that Hudson River water temperatures track air temperatures and the rate of Hudson River warming has followed global warming trends. Significantly, the most recent years studied showed the greatest prevalence in warming. Ashizawa, D. and Cole, J.J., *Long-Term Temperature Trends of the Hudson River: A Study of the Historical Data*, Estuaries Vol. 17, No. 1B, p. 166-171 (March 1994).

Another factor which has relevance to the operation of Indian Point is the fact that the Hudson is an estuary river, and, in the area where Indian Point is situated, the constant action of the Atlantic tides mixes salinated water with the freshwater inflow. During extended droughts, the water near Indian Point is thus, very likely to become more salinated, which can accelerate deterioration of the facility's piping, systems and structures.

In addition to potentially affecting the Hudson River's supply of cooling water and the chemistry of the water in the vicinity of Indian Point, drought and extreme heat waves significantly increase the risks of fire, including that of forest fire in the wooded environ surrounding Indian Point.

Scientists have found growing evidence tying an upsurge in wildfires (as well as hard-to-control large fires set purposefully or accidentally by humans) to climate change. In a report released in July 2006, researchers at California's Scripps Institution of Oceanography and the University of Arizona found that forest fires in 11 Western states over the past 34 years had increased in both size and severity. They concluded that global warming was a key factor.

Most significantly, increased incidence of such fires is not only occurring in warm climates and parched regions such as Australia and the American West and Southwest (e.g. the October 2007 fires which raged in

California). Wildfires are occurring even in Siberia Russia's Sukachev Institute of Forestry reported that more than 29 million acres burned in Russia in 2006. and Alaska. In 2002, wildfires burned on and off in the Fairbanks area and further north for months, starting in mid-May of that year. In 2001, two major fires occurred on Alaska's Kenai Peninsula.

Global warming also brings to New York the likelihood of increasingly violent and more frequent storms. The Commissioner of New York City's Department of Environmental Protection, Emily Lloyd, is one official who has noted that intense storms are expected to occur more frequently in coming years and that flooding from such storms presents a risk to underground systems. New York Times *Flood-Soaked Queens Blames Development, Lagging Sewers and Climate Change*, by Ellen Barry (August 29, 2007). The authors of the report, *Confronting Climate Change in the U.S. Northeast: Science, Impacts, and Solutions*, (July 2007) also warned that New York and the Northeast faces increased storm activity with attendant storm surges.

Westchester, where Indian Point is located, has experienced an unprecedented series of bizarre weather events over only the past three years, including tornados, wind storms of extraordinary velocity (e.g., the winter 2006 wind storm) extremely violent lightning storms (e.g., the

lightning storms of July and September 2007), and torrential downpours (e.g., the April 2007 storm which created "once-in-a-century" floods, rendering parts of Westchester designated national emergency zones).

The December 2007 flooding and storm surges which have left river areas in U.S. Pacific Northwest illustrate how quickly and dramatically rivers can overflow.

Storms present a number of risks to Indian Point. Storm surges and flooding can facilitate the corrosion and disintegration of buried piping and other plant components and systems. Repeated flooding and the rise of the water table can also effectuate alterations in the terrain, particularly in the composition and solidity of the ground and soil which forms the foundation for plant structures and systems. Destructive wind and lightning storms can disrupt off-site and on-site electrical power in a wide variety of ways. Lightning storms also present the obvious risk of igniting fires, either initiating on-site or spreading from adjacent forested areas.

Notably, *both* low water levels in the Hudson River and storms significantly increase the risk of clogging of critical piping and intake structures from river debris. The risk presented by clogging was illustrated in February 2007, when Entergy was forced to declare an "unusual event" emergency after leaves and branches clogged an IP3 water intake structure

during a low tide. Entergy acknowledged the amount of debris to be "significant." Entergy was also forced to shut down its FitzPatrick nuclear station in Scriba, in Oswego County, New York in September 2007 when that plant's water intake became clogged by grass and natural debris. Manifestly, major storms could easily create more robust debris resulting in more devastating incidents.

Finally, any expectation that the operator of Indian Point would have ample time to evaluate and respond to climate change, at some unspecified point in the future is misguided. Just this past summer the Artic ice cap melted at a rapidly accelerated pace, much quicker than predicted by even the most pessimistic scientists. The ice-albedo feedback and the rate of acidification of oceans has also been far greater than virtually all models had anticipated. These, and other, trends strongly indicate that destabilizing feedbacks are beginning to occur that may be gaining force quickly. This vicious dynamic not only raises the likelihood that global warming will accelerate, but the likelihood that highly variable, unprecedented and extreme climatic events can strike without warning.

Conclusion: Stakeholders assert that the effects of climate change on the Indian Point facilities constitute an aging phenomenon that

must be adequately managed. Applicant's failure to take the conditions of climate change into account in its aging management plan renders the LRA fatally flawed in that it leaves numerous systems, structures and components inadequately analyzed and unmonitored during the period of extended operation.

Stakeholders further aver that any grant by the NRC of a new superseding license would constitute an abrogation of the Commission's primary duty to ensure adequate protection of public health and safety.

CONTENTION 50: Replacement Options: Stakeholders contend that the energy produced by Indian Point can be replaced without disruptions as the plants reach the expiration dates of their original licenses.

Stakeholders contend that the Environmental Report in the LRA is deficient because the alternatives analysis is seriously flawed. It lacks an objective evaluation of all reasonable alternatives. The energy produced by Indian Point can be replaced without disruptions as the plants reach the expiration dates of their original licenses.

Replacement options for the electricity produced at Indian Point are available today and more can be planned for in an orderly manner without

disruptions to the supply of electricity as the plants reach the expiration date of their original licenses. This has been well documented by both the Levitan Associates Report commissioned by Westchester County and the by National Academy of Sciences and Engineering, National Research Council, Board of Energy and Environmental Systems (NAS), 6/06 (Exhibit RR), *Alternatives to the Indian Point Energy Center for Meeting New York Electric Power Needs,* which states:

> If some part or all of Indian Point is closed a technically feasible replacement strategy would be achievable. There are no insurmountable technical barriers to the replacement of Indian Point's capacity and energy.
>
> A replacement strategy would most likely consist of a portfolio of approaches, including investments in energy efficiency, transmission and new generation.

It is well within the purview of county and state governments to develop an energy portfolio that will more than compensate for the base load electricity generated at the plant and for the market to respond by providing additional generation. Renewable technologies can be brought on-line quickly. Indian Point's retirement will spur new financing for transmission lines of already approved non-nuclear power plants, solar power, geothermal systems, and wind farms and decentralized and new transmission.

Indian Point will have to close. The question is not if, but when. There are other ways to compensate for the electricity Indian Point's produces and to provide for the region's future energy needs without this facility. The development of a sustainable energy portfolio and conservation and efficiency plan for the region will not only enable the closure of Indian Point, but will put New York on the path of a cleaner, safer and more sustainable energy future. (See Replacement Energy for Indian Point – How Much do we Need, Exhibit XX)

Demand Side Options: Energy Efficiency and Conservation

Samuel W. Bodman, the US Secretary of Energy, stated, "We waste 30% of our energy because of inefficient lighting and appliances and inadequate insulation." Moreover, the energy expert Charles Komanoff wrote "Simply by increasing efficiency and turning off phantom power we will reduce our energy needs dramatically." In fact New York could save a minimum of 1,160 MW a year based on California experience.

Legislation can be enacted to require all municipal and public buildings to turn off unnecessary energy systems when not in use, and have internal monitoring of temperature control systems. New public structures can be required to incorporate green construction principles into their

building plans. This will reduce power consumption, reduce pollution, and save taxpayer monies over the long term.

It is critical to educate the public about wasteful, expensive practices and create incentives that would require that all appliances be shut off when not in use. Buildings and appliances can be required to have easy on/off switches. Once people are aware of the cost of phantom power, they will voluntarily choose to reduce their consumption.

Demand side options represent the cleanest and cheapest form of electricity replacement. Reducing peak loads is far more economical than the cost of installing additional capacity and is already being done across the country. A well thought out energy policy incorporates a portfolio of specific numbers of saved megawatts and lists how goals will be achieved.

In New York, NYSERDA has three programs already in effect:

- The Peak Load Reduction Program which is expected to conserve 355 to 375 MW annually;
- Enabling Technology for Price Sensitive Load Management which is expected to avoid the need for 308 MW
- Keep Cool Program which anticipates a 38 to 45 MW savings;

These programs have saved approximately 700 MW and illustrate how demand side options can reduce peak demand. Reducing peak demand means that generating capacity and reserve margins can both be reduced. Thus, according to the 2006 National Academy of Science study, investments in reducing peak demand through energy efficiency measures can be valued at 118 percent of the actual reduction in megawatts because it avoids the addition of new generating capacity with all its attendant costs. Consolidated Edison has established several demand management programs with the goal of reducing peak load growth by 535 MW; these programs use energy efficiency, smart equipment choices, load reductions programs and distributed generation. The New York Power Authority has committed $100 million a year for energy efficiency projects. These projects and a review of the plethora of alternatives to Indian Point are detailed in the New York State Notice of Intention to Participate and Petition to Intervene and Supporting Declarations and Exhibits filed with the NRC by the New York Attorney General's Office on November 30, 2007 (NY Intervenor Contentions) (Exhibit HH). Stakeholders fully support the NY Intervenor Contentions and incorporate same by reference, as if fully set forth herein.

One way to conceptualize and create a market for demand side electricity savings is via negawatts. Negawatt power is a way of supplying additional

electrical energy to consumers without increased generation capacity. The creation of markets for the trading of negawatts leads to increased efficiency. The concept was introduced by energy expert Amory Lovins, Director of the Rocky Mountain Institute. He first used the term in a 1989 and it has proven to be an efficient measure of saved electricity. The concept works by utilizing consumption efficiency to increase available market supply rather than by increasing plant generation capacity. For example an industrial consumer can advertise for bids for 100 MW hours. A supplier may find energy efficiencies within an unrelated business and contract to improve their heating or lighting for instance. The savings, or negawatts, can then be sold through the utility to the industrial consumer. This becomes an arbitrage transaction rather than an in house process and does not require increased generating capacity from the electric power utility. Entrepreneurial forces are focused on making money by selling negawatts, or saved units of electricity, and the entire system benefits.

Energy consumers may also reduce energy consumption for a few hours to "generate" negawatts. For example, by shutting off air conditioners for a few minutes on the hour, a lot of energy can be saved over a short period of time. Con Ed has already initiated a program for customers in Westchester

which provides a programmable thermostat for air conditioners. The installation is free and the customer receives a stipend. In return they allow Con Ed to turn off their air conditioner for five minutes on the hour a limited number of times daily should electricity supplies run low during peak demand times. In this case the utility is producing and transferring the negawatts. In an expanded market many other vendors could do the same thing and the basic infrastructure remains unchanged. This is a practical and efficient way to get more work done with less electricity without building additional base load generating capacity to replace Indian Point.

Better price signals to the consumer, such as off peak discounts for electricity usage, could change the load profile and allow a better pairing of demand to capacity. One example of this is using discounted off peak pricing to encourage people to shift the time for energy intensive household chores such as washing and drying laundry. On a system-wide basis the shift could be significant. It would also reduce the overall cost of electricity because peak power is more expensive than average costs.

Experiments have been done with meters inside the home which measure the amount of electricity used by household appliances as they are running. The results clearly indicated that when consumers become more aware of

how much electricity is being used and where it is being used, they took steps to reduce usage. Electricity is invisible to most consumers. Making it more visible, that is, giving people information about how to save electricity and making it worth their while to do, can so can result in significant savings. A bill currently pending in the New York State Legislature, (Number A8739) would amend the public service law, in relation to providing real time smart metering technology to residential electricity customers. The purpose of the bill is to facilitate the use of smart metering so that households can reduce the cost of electrical services. It would help consumers reduce the peak demand for electricity.

The experience in California validates this point and illustrates that a 15% reduction in electricity usage can be achieved. The fact that Vermont has held its energy use constant while expanding its economy is proof that this can be done in the East Coast. In many ways it is the community mindset that establishes the parameters of what is possible. In Burlington even hotel guests are expected to recycle. They are also given the option of not having their sheets changed every day in order to save the electricity used in washing and drying them. While public education campaign is needed to reach this mind set, the potential energy savings are huge.

Solar

Solar power is particularly important because it generates the most power during the summer months' peak use period.

New York City has 154,000 sq. acres (about 6.9 billion sq. feet) of available roofs, which could produce 2200 MW. Approximately 7% of this rooftops area is public space (10,780 acres). Approximately 8% of that is municipal/commercial and/or industrial space (12,320), which alone could produce 600 MW. A 2007 study out of the University of Albany also found that 2000 MW could be generated by simply installing solar installations in available parking lot space.

Other solar technologies are being developed, including the Stirling engine and nanosolar, during the next few years solar efficiencies will continue to increase, as the costs continue to decrease.

It was reported on March 16, 2007 in Energy and Capital that DOE made a commitment of $168 million through 2009 for R&D in Solar technologies. The goal is to achieve cost parity with grid electricity by 2015, and to build new "zero energy" homes (homes that produce all of their own power). The program aims to accomplish this by increasing the U.S. manufacturing capacity of solar equipment more than tenfold in the next

three years – a breathtaking rate of build out – which should drive the cost per kilowatt-hour down from about 23 cents to between 5 and 10 cents, about the cost of grid power today.

Wind

New York State has an abundance of wind power upstate. Developing transmission lines, such as the Empire Connection Transmission Lines under the New York State Thruway would quickly bring energy into the Hudson Valley region. (See www.synapse-energy.com.)

A 50 MW wind farm can be completed in 18 months to two years. In addition Mini-Roof Wind Turbines of 3 KW each, at only 15 mph, on 1000 rooftops in any city, would be the equivalent of a 3 MW wind farm, without the associated costs of grid connection. Each unit has a footprint of less than 300 square feet. www.newelectricityworks.com

Geothermal

Geothermal heat pump systems, also known as "geoexchange," are the most energy-efficient, environmentally clean, and cost-effective space conditioning systems available, according to the Environmental Protection Agency. For every 100,000 units of typically sized residential geothermal

heat pumps installed, more than 37.5 trillion BTU's of energy used for space conditioning and water heating can be saved, corresponding to an emissions reduction of about 2.18 million metric tons of carbon equivalents, and cost savings to consumers of about $750 million over the 20-year-life of the equipment. Geothermal heat pumps strengthen U.S. energy security. Every 100,000 homes with geothermal heat pump systems reduce foreign oil consumption by 2.15 million barrels annually and reduce electricity consumption by 799 million kilowatt hours annually. Geothermal heat pumps are efficient. The use of geoexchange lowers electricity demand by approximately 1 kW per ton of capacity.

www.geoexchange.org/documents/GB-003.pdf

Using the constant temperature of the earth of 54 to 55 degrees Fahrenheit to condition interior temperatures, geothermal is available in many areas and is especially helpful for air conditioning, when the use of energy is increased. Additionally instead of heating air that may be 32 degrees or below, with geothermal the air only needs to be heated from 54 to 70 degrees, which greatly reduces energy needs. Geothermal heat pumps are also highly efficient. The use of geothermal exchange lowers electricity demand by approximately 1 KW per ton of capacity.

Modern hydro-power, including wave and tidal power.

Other sustainable technologies such as tidal may be developed in the New York metropolitan region. For instance, Indian Point is located on the Hudson, which is a tidal river.

Creating a Workable NY Energy Market Structure

The 2006 National Academy of Science study on replacing Indian Point's power revealed the excessive complexity and essential dysfunctionality of New York's post-deregulation energy market structure. (See Exhibit RR.) The NAS noted that, in NY, there is no clear nexus between the costs of various forms of electric generation and the ultimate rates paid by consumers. The panel determined that, regardless of whether Indian Point remained on line for another 20 years, if the state wants to avoid serious reliability problems in the future, the flawed energy market needs to be restructured.

One of the most obvious and serious problems, is that the current market structure creates a strong financial disincentive for conservation and efficiency. As things now stand, electricity generators and distributors make

profits by selling more and more electricity. Their strong commercial interest thus is to encourage the public into excessive energy use and waste.

The NAS characterizes Indian Point as a "price taker, not a price setter. This means that the system frequently allows the electricity from Indian Point to be sold at a premium over and above the cost of production. While this maximizes revenue for Entergy, it does nothing to encourage conservation or to lower the cost of electricity being sold to providers. Utilities make their money by pushing consumption not conservation and the Public Service Commission is now looking at alternative business models that would be more responsive to today's circumstances.

Alternatives Offer Significant Cost Savings and Benefits to New York

New York has long suffered the burdens of environmental degradation left behind by polluting industries. History shows that the polluter, Entergy, will be passing the burden of massive quantities of high and low level radioactive waste to the New York taxpayer. Other states, recognizing this reality, have passed legislation charging nuclear power plant operators for dry cask storage, and using the resulting funds to create renewable energy. In 1994, for example, the Minnesota Legislature passed a law to create a renewable energy fund by charging the owner of the Prairie Island nuclear plant for dry

cask storage at the rate of $500,000 per cask, per year. The law was updated in 2003 to set the charge at $16 million per year.

Since a nuclear waste storage facility was not initially approved by New York, the new use of the land for dry cask storage may be taxed accordingly, with the revenue being used to support NYSERDA programs and incentives.

Policy makers and consumers in our region are increasingly aware of the greenhouse gases associated with the production of electricity. Substantial quantities of such gases are produced as part of the full nuclear fuel cycle. Studies show that nuclear energy produces approximately the same amount of greenhouse gases as natural gas (and that the amount will increase fairly soon, as the high quality uranium ore reserves are depleted). Nuclear is not a sustainable form of power. In a carbon constrained world, renewables such as wind, solar, tidal and geothermal power represent the technologies which will produce far less full fuel cycle emissions. These alternatives also do not produce prodigious amounts of highly toxic waste as a by-product.

In sum, the benefits of Indian Point Closure, and a transition to sustainable energy economy include:

- Elimination of the threat of a catastrophic nuclear power accident that could ravage the New York metropolitan region.
- Elimination of one of the most attractive terrorist targets in the nation.
- Reduced production of high level nuclear waste.
- Elimination of the unplanned radiation leaks.
- Elimination of the health-harming radiation released into the environment by Indian Point as part of its normal operation.
- The creation of market incentives to support distributed generation and renewable power.
- Reduced exportation of dollars out of New York State.
- Putting the ownership of the means of electricity production into regional hands.
- New job creation. The number of new jobs will exceed Indian Point jobs (and Indian Point jobs will be phased out gradually, due to the prolonged process of decommissioning).
- Reduction of the global warming gases produced by the entire nuclear fuel cycle.

- Reduction of the cost to the taxpayers of Indian Point (including the safety, security, environmental and public health costs).

Conclusion: Indian Point is an aging generating unit which provides up to 16% of the electricity used within the New York metropolitan region. (See Replacement Energy for Indian Point – How Much do we Need, Exhibit FF.) The electricity can be replaced in many different ways without threatening the stability of the grid. Replacing 2000 MW of base load generation with an equal amount of base load electricity is unnecessary for the integrity of the system if demand side options, supply side options and transmission. (See Replacement Energy for Indian Point – How Much do we Need, Exhibit FF.)

There are large known and unknown costs associated with the waste Indian Point produces. The longer the plant operates, the more radiation and accumulated nuclear waste there will be to deal with. Stakeholders contend that the health and safety of the 21 million people in the metropolitan area requires that the operating license for these two nuclear reactors not be renewed.

CONTENTION 50: Failure to Address Environmental Impacts of Intentional Attacks & Airborne Threats

The Environmental Report ("ER") for the Indian Point 2 and 3 LR is inadequate to satisfy the National Environmental Policy Act ("NEPA") and NRC regulation 10 C.F.R. § 51.45(b) and (c) for the following reasons:

(a) it fails to address the environmental impacts of intentional attacks on the proposed nuclear power plants, or to evaluate a reasonable range of alternatives for avoiding or mitigating those impacts.

(b) it fails to address the cumulative impacts of an intentional attack on the existing Indian Point 1, 2, & 3, or to evaluate a reasonable range of alternatives for avoiding or mitigating those impacts.

c) it fails to address the cumulative impacts the loss of integrity of aging compromised structures, systems and components and a terrorist attack on Indian Point Nuclear Plant.

Basis:

NRC regulations implementing NEPA, 10 C.F.R. §§ 51.45(b) and (c), require Entergy's Environmental Report (ER) to address the impacts of the proposed new 20 year superceding license and operation, on the environment, as well as alternatives for mitigating or avoid those impacts. The ER for the Indian Point 2 and 3 faisl to satisfy these requirements, the

environmental impacts of intentional attacks on the Indian Point during the proposed 20 year new license period.

The NRC's policies and procedures for preparing against terrorist attack, including a "top to bottom" review of NRC security procedures and the establishment of the Office of Nuclear Security and Incident Response, required since 9/11. This is a clear demonstration that the NRC considers such attacks to be reasonably foreseeable for purposes of requiring a NEPA review. *San Luis Obispo Mothers for Peace v. NRC,* 449 F.3d 1016 (9th Cir. 2006) ("*Mothers for Peace*").

On 9/11 one of the hijacked planes used the Hudson River as a navigational guide, flew directly past the twin domes of the Indian Point Reactors. The 9/11 Commission learned that the terrorist's original plan included 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.

The 9/11 Commission's investigation found that when Mohammed Atta was conducting his surveillance flights, he came close to redirecting the strike on Indian Point.

National Research Council analyses and post-9/11 intelligence has also indicated that the U.S. nuclear infrastructure is 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 Nuclear Facilities,* March 27, 2007 Re: *Proposed Rule: Power Reactor Security Requirements* (RIN 3150-AG63)

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

that found

> **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),**

In *Mothers for Peace,* the U.S. Court of Appeals for the Ninth Circuit reversed the NRC's refusal to include the environmental impacts of terrorist attacks in its licensing decisions. *See Pacific Gas & Electric Co.* (Diablo Canyon Independent Spent Fuel Storage Installation),

CLI-03-01, 57 NRC 1 (2003); *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-02-25, 56 NRC 340 (2002).

Pacific Gas & Electric Company ("PG&E") had petitioned the Supreme Court for a writ of certiorari regarding the *Mothers for Peace* decision. However, the NRC's did not even file its own petition for certiorari, or submit a timely response in support of PG&E's petition, thereby indicating that the NRC did not consider the decision to warrant Supreme Court review, and that they accepted Ninth Circuit's mandate.36 [4]

The Supreme Court denied certiorari, early 2007 and making the ruling in Diablo Canyon's "Mother's For Peace" case the Ninth Circuit Court requiring that the effects of a terrorist attack be included in the ER required in licensing proceedings by the NRC. Regulations 10 CFR 51.53 to fulfill the NRC's NEPA requirement, is the final legally binding decision.

NRC own precedent was set in the licensing process for the Irradiation Facility in Hawaii , where the NRC included terrorist threat in its consideration of approving a license

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[4] 36 *See* Sup.Ct.R. 12.6, which provides that:

> All parties other than the petitioner are considered respondents, but any respondent who supports the position of a petitioner shall meet the petitioner's time schedule for filing documents, except that a response supporting the petition shall be filed within 20 days after the case is placed on the docket, and that time will not be extended.

The Supreme Court docketed PG&E's petition for certiorari on October 3, 2006. *See http://www.supremecourtus.gov/docket/06-466.htm.* Pursuant to Sup.Ct.R. 12.6, if the government wished to file a brief in support of the petition it was required to do so by October 23. Therefore it is reasonable to expect that the government's brief, now due on December 15, 2006, will oppose the taking of certiorari.

The NRC argues that since this was a Ninth District Court decision, it does not apply to Nuclear Plants outside the Ninth circuit. However, all United States citizens are afforded equal protection under the law, and therefore the issue of adequate assurance against the efforts of a terrorist attack on a nuclear plant must be equally applied to all nuclear facilities throughout the nation.

Therefore the NRC should apply the same standards to all of its licensing decisions, especially at Indian Point located proximity to New York City 25 miles, West Point Military Academy and a United States Mint are located on 8 miles away.

After 9/11 Homeland Security was created to protect the United States against terrorism.

However, since 9/11 the NRC has only made minor changes in the Design Basis Threat that a licensee must defend against.
Stakeholders primary concern is that the way the force-on-force (FOF) tests are conducted do not prove that the Indian Point security force is capable to defend the facility against a credible terrorist attack or sabotage. The LRA does not address how Security will be managed during the proposed additional 20 years of operation against sabotage/terrorist forces that during

that time will continue to gain sophistication and access to advance weapons.

In a letter dated 12/9/03 by Project on Government Oversight to the NRC then Chairman Diaz, the following issues that must be addressed to insure adequate security were raised:

Dumbed-Down Design Basis Threat (DBT) – It has been widely reported in the press that prior to 9/11, nuclear power plants were required to have defenses designed to protect against only a ridiculously small attacking force – three terrorists. In contrast, the intelligence community generally believes that terrorists would attack a target with a squad-sized force, which in the Army special forces is 12 and the Navy Seals is 14. In other words, the NRC would need to at least quadruple its old DBT.

NRC's new DBT, does not reach the 12 to 14 level that is appropriate. Representatives of other federal agencies have told Project on Government Oversight that the NRC's new DBT remains inadequate. (POGO on Nuclear Security Act, exhibit PP) (CONTENTION #19 is referenced and incorporated fully, as if set forth herein.)

The NRC argues that the new DBT is the largest threat against which a private security force can be expected to defend. This rationale is backwards and conflates two separate considerations – what is the size of the

threat and what should the nuclear power industry be required to do to in the face of such threats.

The NRC policy decision to limit the size of the DBT (under pressure from the NEI and its friends in Congress) was based mainly on its assessment of what is reasonable to ask of a private force.

That approach ignores the most fundamental question: what is the credible threat against the facilities? The size of the DBT must be based on that threat. Furthermore, NRC's justification of its too-low DBT rings hollow, as the Department of Energy (DOE) claims to protect its facilities against twice as many terrorists as the NRC does.

Under Use of Readily-Available Lethal Weapons – It is well known in security circles that there are weapons that are available to terrorists that can penetrate bullet-resistant enclosures (BREs), which are quasi-guard towers. BREs are included in the defensive strategy of a number of nuclear power plants, including Indian Point. Some time ago, the Department of Energy abandoned the use of its state-of-the-art guard towers (which are far more robust than most BREs) because of their vulnerability to readily-available weapons. Indian Point officers have been aware of the controversies surrounding BREs and have brought their concerns not only to Entergy, but also to the NRC Region I, with no response at all.

Several years ago, the DOE developed a classified official Adversary Capabilities List which includes weapons and explosives that are readily available to terrorist groups. The NRC should review this list and ensure its Design Basis Threat includes them. For example, .50 caliber sniper rifles (which have been available since World War I) and Armor-Piercing Incendiary rounds (which are available in gun shops for $1 per round) made the DOE guard towers so vulnerable they were abandoned. Other weapons were also of concern, including the rocket-propelled grenades which have been used frequently by near-children around the world in war-torn countries, with great success against hardened targets, and which could easily be launched from boats on the river, or from unpatrolled shores.

Unrealistic Timing and Location of Attack – In 2003, the NRC conducted the three FOF tests at Indian Point during the daylight at the beginning of the night shift, and began at least two of the tests in the owner-controlled area. There are several problems with this:

* The security force being tested had just come on duty and was not yet fatigued by a 12-hour shift, hours typically worked by Indian Point security officers five to six days a week.

* The security officers knew within the hour that the test was to begin, as the day shift was held over an extra hour to cover as a shadow force so that the night shift could be tested at the beginning of their shift.

* No Surprise. The security force knew for months in advance that this test was going to occur, training specifically for the approved scenarios. They even knew within minutes that the test was to occur, because of all the visiting dignitaries and the fact that they had strapped on Multiple Integrated Laser Engagement System (MILES) equipment.

* The FOF tests are not conducted at high speed because of the overriding safety concerns. Therefore, people and vehicles are not going full tilt the way they would during a real terrorist attack, giving the protective forces time to pause to make decisions – time that they wouldn't have in a real life situation. Safety was also used as the reason for not conducting the tests at night. Sources told us that Entergy was worried participants could trip over rocks or step on snakes.

* No Trained Adversaries. The mock terrorists were security officers from another nuclear plant who had no training as adversaries. This training is critically important because it teaches the mock terrorist how to think and act offensively, as a real terrorist would, rather than defensively as a security

guard would. Here again, both DOE and the military use trained adversaries to test their security forces.

Very few of the above issues have been addressed since POGO first put the NRC on notice in 2003, and the LRA does not address any aging management plans with regard to enhancing the security force during the proposed 20 year superceding license.

The world is not becoming a safer place, but a much more dangerous one. Stakeholder contend that the NRC cannot approve a superceding 20 year license for Indian Point 2 or Indian Point 3 without first requiring the Applicant to substantially improve security and develop an aging management plan to maintain such security.

B. INCREASE IN AIRBORNE THREATS

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.

In a 2005 updated, report by Carl Behrens and Mark Holt, Nuclear Power Plants: Vulnerability to Terrorist Attack "Protection of nuclear power plants from land-based assaults, deliberate aircraft crashes, and other

terrorist acts has been a heightened national priority since the attacks of September 11, 2001. the industry has been too slow and that further measures are needed.

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.

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-demonstrate that the NRC considers such attacks to be reasonably foreseeable for purposes of requiring a NEPA review.

The residents in the Hudson Valley, specifically Rockland County, all of which is within 20 miles of Indian Point, have been recently advised of the FAA's decision to increase air traffic in the region by 600 flights a day.. On average every two to three minutes the noise of aircraft flying overhead will be heard.

In Entergy's LRA Environmental Report. ["2.13 Related Federal Project Activities states that Entergy "does not identify any known or reasonably foreseeable federal projects or other activities that could contribute to the cumulative environmental impacts of license renewal at thesite.http://www.nrc.gov/reactors/operating/licensing/renewal/applications/indian-point/1-ipec-lra-appendix-e_1- 2.pdf, pg 113 of 156).

However, this statement is incorrect. The FAA Redesign Project is a Federal project that has been considered since 1999, the increased background noise due to increased air traffic might have on the efficacy of the emergency alert system.

The new alarm system, whose incomplete installation demonstrates Entergy's inability to meet a Federal law deadline, cannot be heard inside a house or even in a parked car.

There is no no-fly zone over Indian Point. This presents a clear and significant danger since planes of all shapes and size, including private jets and large commercial planes. There are at least 7 major airports within the 50 miles of Indian Point, including Westchester County Airport, Stewart International Airport, JFK International Airport, La Guardia Airport, and Newark International.

International carriers are planning to use the plane for flights in and out of Kennedy,. In January 2008, Airbus will be flying into Steward Aiport, located

approximately 9 miles from Indian Point. Aribus's superjumbo A380, the world's largest passenger plane, It has **a** wingspan almost as long as a football field, it is eight stories tall, and it weighs 118 tons heavier than the Boeing 747, the planes that were used in the terrorist attack on 9/11.

> the biggest purchases of Airbus are from the United Arab Emirates., the craft is certified to carry up to 853 — about twice the capacity of the biggest version of the Boeing 747. (March 2007 NYT).

OTHER THREATS

Also, 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.

Additionally Indian Point's vulnerable location, is evidenced by the fact that it located across from Tompkins Cove, approximately less than a mile across the Hudson, where there is a large unpatrolled easily accessible area, where cargo trains run multiple times a day.

Additionally the danger of an internally set fire or fires is very real. (CONTENTIONS #4 - #11 are referenced and incorporated fully, as if set forth herein.)

The risk of a terrorist attack on a nuclear reactor site is a very real possibility. Therefore the ER must provide a full discussion of the potential consequences of a range of credible events involving destructive acts against the proposed reactors. The range of events considered in the ER should include all types of attacks that are reasonably foreseeable..... *Limerick Ecology Action v. NRC,* 869 F.2d 719, 726 (3rd Cir. 1989).

MATERIAL FACTS & LAW

The Stakeholders have made "a minimal showing that the material facts are in dispute, thereby demonstrating that an inquiry in depth is appropriate. regarding to matters of both law and fact.

If the NRC approves this proposed new superseding 20 year operator's license to the Applicant, it will be evidence that the NRC really takes its marching orders from the NEI, rather than from the Laws of the United States. There is no justification for permitting national security to be compromised – even a little – to safeguard corporate profits. The

NRC's organizing mandate is to adequately protect public health and safety. It the NRC does not require the realistic threat of terrorism to be considered in the herein new license application, it violates NEPA.

It is against public policy to allow health and safety cannot be grandfather-in, only seven years after 9/11. Therefore Stakeholders contend that the Applicant LRA must be denied, because it does not address the impacts of a terrorist attack.

CONTENTION 51: Withholding of Access Proprietary Documents Impedes Stakeholders Adequate Review of Entergy Application for License Renewal of IP2 LLC and IP3 LLC.

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

10 CFR Rules and Regulations defines and spells 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. For contentions to be accepted they must meet a minimal standard of proof in raising a contention a matter of law or fact, 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.

Despite the additional time for Stakeholders to file Petitions to intervene and request for hearings, due to document access issues, caused by delays of other governmental agencies and malfunction of ADAMS, the NRC's liberal granting of proprietary status to nuclear industry documents, including massive redactions [on the claim of proprietary information] inferred with Stakeholder's ability to review the LRA documents support their contentions. Conversely Entergy had unlimited access to all materials and reports, as well as, years and nearly unlimited resources to prepared the LRA.

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.

Stakeholders contend that 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 leak 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) Stakeholders contend that it is imperative in reviewing the proposed LRA, that Stakeholders be give a fair opportunity to measure the reliability and adequacy of the aging management plans contained therein by having access to all relevant documents. Anything less creates a bias in favor of the Applicant.

(ii) In measuring the potential adverse effects associated with the proposed action (license renewal) the Stakeholders have done due diligence in reviewing available information. Citizen volunteers, attorneys and our industry expert have dedicated thousands of 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 being fully able to support certain contentions.

(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, 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, 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.

Other examples revolve around industry documents that the Applicant relies upon in the formulation of its Aging Management Plans that

are not available for review under the same proprietary claims. Stakeholders know of issues regarding Boraflex degradation 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. 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.

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 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. Stakeholders rights 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 the Stakeholders access to all relevant information based on the Applicant's claims to proprietary privilege may cause irreversible harm to the Stakeholders and the Stakeholders' community.

A community and its citizens' right to be involved in the licensing process is within scope, and is 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 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, that relate to the NRC considers 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 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.

The NRC, in their method of conducting a License Renewal Process, has designed it, upon the assistance of the Nuclear Energy Institute (NEI), to eliminate meaningful citizen involvement, and have attempted to thwart the right of redress, as is guaranteed by the Constitution and Bill of Rights.

By the Applicant hiding crucial documents behind the veil of Proprietary Privilege, and the NRC's liberal granting of proprietary privilege creates a 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.

A document that was very relevant to Contentions # 22- 26 was requested on September 25, 2007. It took more than two months to receive, it because of an apparent need to review the document for proprietary information. After losing confidence in the NRC in producing the document prior to the November 30th deadline. Our expert witness found it through other resources, because it was in the public domain, and unredacted.

The NRC took two months to redact a version which they held back until the last week in November. However, when the copy we received was compared to the redacted version, it appear there was no reason, except for interference with Stakeholders right to documents relevant to the LRA, that it was delayed and redacted.

Essentially, our of 4000 pages. Figures 1.2-5 thru 1.2-9, Figures 5.1-2 thru 5.1-6, Figures 6.2-2 thru 6.2-3, Figure 7.7-1, Figures 10-7, 10-7a, Figures 11.2-3, Figure 5-16 of Appendix 14a, Figure Q9.5-1

The "so called" proprietary material was illegible to begin with. Frankly totally unusable, and it is doubtful that any of it meets the requirements for withholding this information, under Part 2 rules.

The Applicant has deliberately and knowingly caused another person (NRC Staff) to withhold documents from and official proceeding (License Renewal Application Process). The Applicant's over use and misuse 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 liberal granting of said privilege without question makes both the NRC and Entergy complicit in attempting to withhold and 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."

The NRC is required 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.

By granting liberal proprietary privileged status to a plethora pertinent documents, the NRC failed in their fiduciary duties and responsibilities the

public. Instead of making decisions 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 almost ALL requests by the Applicant for Proprietary Privilege. NRC's in-house protocols, in this regard, violate own regulations, and have placed members of the public at a disadvantage by interfering with the Stakeholders' right to redress.

It regard to 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 and outweigh the NRC's desire to keep to a tight time schedule in the relicensing process.

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.

Stakeholders contend that the Applicant's claim to Proprietary Privilege has adversely affected Stakeholder's ability to fully review the LRA and submitted supported contentions. Therefore until Stakeholders have fair and adequate access to records and documents the NRC should not close the time period in which Stakeholders can submit Petitions to Intervene with Contentions and Request for Hearings. Stakeholder contend that Stakeholders are being unfairly barred from being able to adequately present support contention, and that the LRA has been prejudiced in favor of the Applicant.

V. CONCLUSION AND REQUEST FOR RELIEF

Stakeholders support, reference and incorporate fully, as set for herein New York State Notice of Intention to Participate and Petition to Intervene and Supporting Declarations and Exhibits filed with the NRC by the New York Attorney General's Office on November 30, 2007 (NY

Intervenor Contentions). Further Stakeholders reserve their right to amend this Petition, without prejudice.

The above 51 CONTENTIONS were prepared by a group of concerned Citizens living in the Hudson Valley. The Depth and Breath of the problems we assert above are just the tip of the iceberg.

Our research has shown clear evidence that the components, systems, and structures at Indian Point 2 and Indian Point3 are corroded with age. They have lost their structural integrity and are leaking. They can no longer do the jobs they were designed for. It would be negligent for the NRC to approve Entergy's License Renewal Application.

Stakeholders are formally placing the NRC on NOTICE that the problems raised in the CONTENTIONS will exist during the proposed new 20 year license period. Many exist today.

The NRC has an obligation and a mandate to "Adequately Protect Public Health and Safety", not corporate profits. Stakeholders

implore the NRC Staff and Commissioners to do their job, as a regulators.

The NRC's License Renewal Process is onerous and obstructionist, to the extent of being almost punitive to citizens Stakeholders. The seemingly systematic withholding of information, and limited time afforded communities to prepare Petitions to Intervene seems to be deliberately biased in favor the nuclear industry.

(i) Entergy has had years to prepare its License Renewal Applications.

While, Stakeholders had 120 days to research and write Petitions to Intervene with Contentions, during which time ADAMS malfunctioned for nearly a month.

(ii) Entergy had unfettered access to documents.

While, Stakeholders had severely limited access to relevant documents, many of which were purposely withheld.

(iii) Entergy makes half a billion profit annually, and is paying for the top law firm and PR firms.

While, Stakeholders are volunteer citizens with limited resources, whose sole motivation in submitting the above CONTENTIONS is a heartfelt effort to protect the safety of our community.

Our lives, our children, our families, our homes, our neighbors, and our futures are at stake.

Therefore,

We respectfully request that all the Contentions raised in this Petition be fully accepted for review by the ASLB.

We respectfully request a public HEARING on all the Contentions raised in this Petition.

We respectfully request that any and all HEARINGS regarding the herein LRA be held within the Hudson Valley accessible to Stakeholders.

We respectfully request that the NRC deny Entergy's LRA as being incomplete, inaccurate, unacceptable and unfixable.

We respectfully request that due to the incomplete and inaccurate LRA submitted by Entergy, that the NRC reject Entergy's LRA, as null and void.

December 10, 2007 WESTCHESTER CITIZEN'S AWARENESS NETWORK; ROCKLAND COUNTY CONSERVATION ASSOCIATION; PUBLIC HEALTH AND SUSTAINABLE ENERGY; SIERRA CLUB-ATLANTIC CHAPTER; and NEW YORK STATE ASSEMBLYMAN RICHARD BRODSKY

Susan H. Shapiro, Esq.

Assemblyman Richard L. Brodsky

Representing:
Westchester Citizen's Awareness Network
Rockland County Conservation Association
Public Health and Sustainable Energy
Sierra Club – Atlantic Chapter
New York State Assemblyman Richard Brodsky

We respectfully request that the NRC deny Entergy's LRA as being incomplete, inaccurate, unacceptable and unfixable, and in violation of applicable law and regulation.

December 10, 2007 WESTCHESTER CITIZEN'S AWARENESS NETWORK; ROCKLAND COUNTY CONSERVATION ASSOCIATION; PUBLIC HEALTH AND SUSTAINABLE ENERGY; SIERRA CLUB-ATLANTIC CHAPTER; and NEW YORK STATE ASSEMBLYMAN RICHARD BRODSKY

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Representing:
Westchester Citizen's Awareness Network
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Public Health and Sustainable Energy
Sierra Club – Atlantic Chapter
New York State Assemblyman Richard Brodsky

WestCAN, RCCA, PHASE, SIERRA CLUB and
ASSEMBLYMAN RICHARD BRODKSY

PETITION TO INTERVENE with CONTENTIONS & REQUEST FOR A HEARING RE: LICENSE RENEWAL APPLICATION FOR INDIAN POINT 2 and INDIAN POINT 3

TABLE OF CONTENTIONS

CONTENTION 5: The Fire Protection Program described in the Current License Basis Documents including the unlawfully approved exemptions to Appendix R, the Safety Evaluation and the amended license for Indian Point 3 fail to adequately protect the health and safety of the public, and fail to meet the requirements of 10 CFR 50 and Appendix R

CONTENTION 6: Fire Protection Design Basis Threat. The Applicant's License Renewal Application fails to meet the requirements of 10 CFR54.4 "Scope," and fails to implement the requirements of the Energy Policy Act of 2005.

CONTENTION #7: Fire initiated by a light airplane strike risks penetrating vulnerable structures.

CONTENTION 8: The NRC improperly rushed approval Entergy's modified exemption request reducing fire protection standards from 1 hour to 24 minutes while deferring necessary design modifications.

CONTENTION 9: In violation of promises made to Congress the NRC did not correct deficiencies in fire protection, and instead have reduced fire protection by relying on manual actions to save essential equipment.

CONTENTION 10: (Unit 2) Cable separation for Unit 2 is non-compliant, fails to meet separation criteria and fails to meet Appendix R criteria. This has been a known issue since 1976; and again in 1984, yet remains non-compliant today.

CONTENTION 11A (Unit 2 and Unit 3): The Fire protection program as described on page B-47 of the Appendix B of the Applicant's LRA does not include fire wrap or cable insulation as part of its aging management program.

CONTENTION 11B: Environmental Impact of an increase in risk of fire damage due to degraded cable insulation is not considered thus the Applicants' LRA is incomplete and inaccurate, and the Safety Evaluation supporting the SAMA analysis is incorrect.

CONTENTION 12: Entergy either does not have, or has unlawfully failed to provide the Current License Basis' (CLB) for Indian Point 2 and 3, accordingly the NRC must license renewal.

CONTENTION 13: The LRA is incomplete and should be dismissed, because it fails to present a Time Limiting Aging Analysis and an Adequate Aging Management Plan, and instead makes vague commitments to manage the aging of the plant at uncertain dates in the future, thereby making the LRA a meaningless and voidable "agreement to agree."

CONTENTION 14: 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."

CONTENTION 15: Regulations provides that in the event the NRC approves the LRA, then old license is retired, and a new superseding license will be issued, as a matter of law § 54.31. Therefore all citing criteria for a new license must be fully considered including population density, emergency plans and seismology.

CONTENTION 16: An Updated Seismic Analysis for Indian Point must be Conducted and Applicant must Demonstrate that Indian Point can avoid or mitigate a large earthquake. Indian Point Sits Nearly on Top of the Intersection of Two Major Earthquake belts,

CONTENTION 17: The population density within the 50 mile Ingestion Pathway EPZ of Indian Point is over 21 million, the population within in the 10 mile plume exposure pathway EPZ exceeds 500,000.

CONTENTION #18 Emergency Plans and evacuation plans for the four counties, surrounding are inadequate to protect public health and safety, due to limited road infrastructure, increased traffic and poor communications.

CONTENTION: #19 Security Plans Stakeholders contend that the way the force-on-force (FOF) tests are conducted do not prove that the Indian Point security force is capable to defend the facility against a credible terrorist attack or sabotage. The LRA does not address how Security, as required under section 10 CFR 100.12(f) and 10 CFR Part 73, will be managed during the proposed additional 20 years of operation against sabotage/terrorist forces with increasing access to sophisticated and advance weapons.

CONTENTION # 20: The LRA does not satisfy the NRC's underlying mandate of Reasonable Assurance of Adequate Protection of Public Health and Safety.

CONTENTIONS 22-26 Indian Point was not required to comply with federally approved General Design Criteria, which constitutes a clear and flagrant violation of the Administrative Procedures Act, and Entergy's LRA fails to remediate the error, leaving Indian Point without adequate safety margins and the New York Metropolitan region without adequate assurance of protection of public health and safety

CONTENTION # 27: The LRA for Indian Point 2 & Indian Point 3 is insufficient in managing the environmental Equipment Qualification required by federal rules mandated that are required to mitigate numerous design basis accidents to avoid a reactor core melt.

Contention #28 - 32 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.

CONTENTION #33: The EIS Supplemental Site Specific Report of the LRA is misleading and incomplete because it fails to include Refurbishment plans meeting the mandates of NEPA, 10 CFR 51.53 post-construction environmental reports and of 10 CFR 51.21. Issue Summary.

CONTENTION #34: Stakeholders contend that accidents involving the breakdown of certain in scope parts, components and systems are not adequately addressed Entergy's LRA for Indian Point 2 and Indian Point 3.

CONTENTION 35: Leak-Before-Break analysis is unreliable for welds associated with high energy line piping containing certain alloys at Indian Point 2 & Indian & Indian Pont 3.

CONTENTION #36: 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.

CONTENTION 37: The LRA and the UFSAR's for Indian Point inadequately address the currently existing (known and unknown) environmental affects and aging degradation issues of ongoing leaks, and fail to lay out workable aging management plans for leaks and critical safety systems

CONTENTION #38: Microbial action potentially threatens all the stainless steel components, pipes, filters and valves at Indian Point (issue 99 of EIS)

CONTENTION #39: Indian Point 1 leaks constitute a violation of SafeStor and since components of IP1 are used in the operation of Indian Point 2, the LRA's failure to address these leaks and the interfacing IP 1-IP2 systems renders the LRA inaccurate, incomplete, and invalid

CONTENTION 40 : 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."

CONTENTION # 41 : Entergy's high level, long-term or permanent, nuclear waste dump on the bank of the Hudson River.

CONTENTION # 42: Dry Cask Storage (Issue 83)
The Independent Spent Fuel Storage Installation (SFSI) being constructed at Indian Point for the purpose of holding the overflow of nuclear waste on site for decades, and probably more than a century, must be fully delineated and addressed in the aging management plan and, moreover constitutes an independent licensing issue.

CONTENTION 43: The closure of Barnwell will turn Indian Point into a low level radioactive waste storage facility, a reality the GEIS utterly fails to address, and a fact which warrants independent application with public comment and regulatory review.

CONTENTION 44 : 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.

CONTENTION 45 Non-Compliance with NYS DEC Law – Closed Cycle Cooling "Best Technology Available" Surface Water Quality, Hydrology and Use (for all plants)

CONTENTION 46: OMIT

CONTENTION 47: The Environmental Report Fails to Consider the Higher than Average Cancer Rates and Other Health Impacts in Four Counties Surrounding Indian Point.

CONTENTION 48 : Environmental Justice - Corporate Welfare

CONTENTION 49: Applicant's LRA fails to consider the effects of global warming and Applicant has failed to present a plan for how it will either analyze or manage such effects during an additional 20 years of operation.

CONTENTION 50: Failure to Address Environmental Impacts of Intentional Attacks & Airborne Threats

CONTENTION 51: Withholding of Access Proprietary Documents Impedes Stakeholders Adequate Review of Entergy Application for License Renewal of IP2 LLC and IP3 LLC.

Appendix G Dept. of Interior, Office of the Secretary Letter p112-p115

Appendix H Consolidated Edison Financial Analysis p116-

Exhibit K

Safety Evaluation by the Division of Reactor Licensing Unit 2, Nov 16, 1970
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Exhibit K Supplement
AEC Regulatory Staff Safety Evaluation IP 2 Nov. 20, 1970 p1-23

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Reactor Coolant System Operational Leakage

Amendment 251

Amendment 238

Exhibit M

Chilk Memorandum Sept 18, 1992 Deviations Identified During SEP

Exhibit N

Criterion 43 - 46 Core Cooling

Exhibit O

Omitted

Exhibit P

Indian Point Licenses Renewal Karl Jacobs

Exhibit Q.1

First Declaration of Ulrich Witte with Summary

Exhibit R

Leak Found in Pipe at Indian Point with graphic

Exhibit S

Kansler Letter Application for Order Approving Indirect Transfer of Control Licenses

Application for Order Approving Indirect Transfer of Control of Licenses p11

Organization Flow Chart p12

Figure 2 Organization Chart - Post Reorganization

Attachment 1 General Corporate information

Attachment 2 Projected Balance Sheet

Attachment 3 Projected Income Statements, Non-Proprietary Version

Attachment 4 Affidavit of Michael Kansler

Attachment 5 Form of Support Agreement

Exhibit U
Omitted

Exhibit V
Synapse Financial Insecurity p1-40

Exhibit W
Corp Watch: Entergy Holds New Orleans Hostage

EXHIBIT X
Memorandum to Chairman Diaz April 25, 2003 NRC Enforcement of Regulatory Requirements . NRC Enforcement Regulatory Requirements and Commitments at Indian Point, Unit 2

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EXHIBIT Y
Status of Decommissioning Funding for Plants Operated by Entergy Nuclear Operations, March 25, 2005. 10CFR50.7a5(f)(1)

EXHIBIT Z
Federal Register, August 1, 2007 Notice of Docketing of Application for Renewal.

EXHIBIT AA
H.R. 994: To Require the NRC to conduct an ISA

EXHIBIT BB
Notice of Availability of the Final License Renewal Interim Staff Guidance and Staff Guidance for Preparing Severe Accident Mitigation Alternatives Analysis. Aug. 14, 2007

EXHIBIT CC

NRC Regulatory Issue Summary 2003-09 Environmental Qualification of Low-Voltage Instrumentation and Control Cables. May 2, 2003

EXHIBIT DD

Entergy Replacement Reactor Vessel Head. Doosan Heavy Industries Co. Ltd.

EXHIBIT EE

CIECP: NRC Proposed Rule: Power Reactor Security Requirements 3/27/07

CRS Report for Congress- Nuclear Power Plant Vulnerability to Terrorist Attack

EXHIBIT FF

Replacing the Electricity at Indian Point – November 5, 2007

EXHIBIT GG.1

Declaration of U. Witte: Review of Contention 35: Leak Before Break

Declaration of U. Witte: Review of Contention 14: Safety/Aging Mgmt

EXHIBIT HH

NYS Notice t Intervene in Petition

EXHIBIT II.2

Declaration of U. Witte: Review of Contention 35: Leak Before Break

EXHIBIT JJ

Entergy Questionairre 7/31/06: Groundwater Protection Baseline Info IPEC 1,2, & 3

EXHIBIT KK

Indian Point 2 2Q/2007 Plant Inspection Findings

EXHIBIT LL

Declaration of Dr. Gordon Thompson in Support of Petitioner's Concern

EXHIBIT MM

Gordon Thompson: Robust Storage of Spent Nuclear Fuel

EXHIBIT NN

Population Growth in Rockland, Orange, Putnam, Westchester 1960-2006

EXHIBIT OO

James Lee Witt Assoc.-Executive Summary : Review of Emergency Preparedness IP/Millstone, 2003.

EXHIBIT PP
POGO on Nuclear Security Act

EXHIBIT QQ
Levitan Study Executive Summary

EXHIBIT RR
NAS Report 2006: Replacement Energy of Indian Point
Executive Summary

EXHIBIT SS
Notes from 8/30/07 conference call regarding Dry Cask Storage at IP

EXHIBIT TT
Public Health Risks of Extending Licenses of the IP2 and 3 Nuclear Reactors, 10/6/07 Dr. Joe Mangano, Radiation and Public Health Project.0

EXHIBIT UU
Expert Witness - J. Mangano - Declaration

EXHIBIT VV
Quality Control – Whistleblower letter

EXHIBIT WW
GAO Report – NRC Liability Insurance

EXHIBIT XX
Replacement Energy for Indian Point: How Much Do We Need?

EXHIBITYY
Blank

EXHIBIT ZZ
Blank

EXHIBIT AAA
Declaration Connie Coker

EXHIBIT BBB
Janet Burnet

EXHIBIT CCC
Andrew Stewart

EXHIBIT DDD
Michel Lee

EXHIBIT FFF
Robert Jones

EXHIBIT GGG
Maureen Ritter

EXHIBIT HHH
New York City Councilman Vaca

EXHIBIT III
Dorice Madronero

EXHIBIT JJJ
Quality Control – Whistleblower Letter

FIRE PROTECTION EXHIBITS 1 -20

EXHIBIT A

Exhibit A

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.		**LicenseNo.**
ENTERGY NUCLEAR INDIAN POINT 3, LLC	)	**DPR-26 &**
ENTERGY NUCLEAR OPERATIONS, LCC		**DPR 64**
Indian Point Energy Center Unit 2 & Indian Point	)	**Docket**
Entergy Center Unit 3		**No. 50-247**
		& No. 50-
		286

License Renewal Application

DECLARATION OF MARILYN ELIE

My name is Marilyn Elie, I live at 2A Adrian Court, Cortlandt Manor, New York, 10567. I live approximately three miles from Indian Point. I am a member and co-founder of Westchester Citizens Awareness Network (WestCAN) and a member of the Steering Committee of the Indian Point Safe Energy Coalition (IPSEC).

WestCAN represents my interests in a Petition to Intervene, Request for Hearing and Contentions and the Notice of Appearance, in the matter of Entergy Nuclear Indian Point 2 LLC and Indian Point 3 LLC, and Entergy Nuclear Operations, Inc. License Renewal Application.

I have lived in the Hudson River Valley since 1975 in locations ranging from New York City to Jefferson Valley, New York. When I decided to move to Cortlandt Manor thirteen years ago the proximity to the Hudson River was a very important consideration. I was born in a Roanoke River Valley and grew up in the Ohio River Valley. I have had a personal and close connection with rivers my entire life. I frequently visit the shores of the Hudson River at various parks up and down the Valley.

I participate in many activities which are centered around the Hudson River such as Earth Day celebrations, Clean Up Days, the seasonal river front

festivals organized by the Beacon Sloop Club from the Strawberry Fest in June through the Pumpkin Fest in the fall. I have attended the Great Hudson River Revival for decades and working for Clearwater is an important part of my life. I sing with Walkabout Clearwater Chorus which sings at festivals up and down the River. I have organized demonstrations at the Peekskill Riverfront Green within sight of Indian Point. I frequent Croton Point Park and the beach there for hiking and bird watching, swimming and wading in the summer. I sail on the Clearwater once a year and on other boats when I can. I enjoy parties, picnics and private celebrations down by the River with friends whose children roam the river bank and splash in the shallows, as I sometimes do myself.

I have been aware of the nuclear reactors at Indian Point for decades and became more involved in learning about the plant and following what happens there in the last twelve years. I became involved in this issue because of my concern about the highly radio active waste which is stored on site. After thirty years, neither the industry or the regulators have a system in place to deal with the highly radio active waste that is a result of generating electricity using nuclear reactors. I have visited Yucca Mountain. Making one state into the nuclear dumping ground for the entire nation is neither moral or practical, especially when the people of the state of Nevada are so opposed to it. This waste has a toxic half life that is longer than it took the dinosaurs to turn into oil. It is not safe to move and not safe to store. It is a deadly legacy to pass on to our children and as teacher working with young children whose future will be impacted, I felt a pressing need to be involved in putting an end to its accumulation.

Other issues at the plant over the years have raised additional grave concerns. These include, but are not limited to, the steam generator leak, current unregulated leaks of radio active water from unknown sources in unknown amounts, the leaking spent fuel pool at Unit 1, regular and routine authorized discharges of noble gases to the air, and the discharge of radio active isotopes such as SR90 to the River which are authorized and considered as below regulatory concern. Dilution is not the answer to pollution.

In addition the plant uses the water in the Hudson River for its cooling system and dumps waste heat back into the River. This is a for profit use of a public resource that is unconscionable. It has altered the ecology of the Hudson River and impacted economic and recreational development along

the River. After years and years of delay the Department of Environmental Conservation has finally taken action and Entergy is contesting the decision.

As I have followed this issue over the years the impact of the reactors on public health and safety, and the environmental has become more and more apparent. It is hard to believe that any other business would be so laxly regulated. The plant imperils millions of people for the benefit of a few. The New York electrical grid is robust, our elected officials are moving aggressively toward conservation, efficiency, and green alternatives. We can replace the power; we cannot replace the people should there be a catastrophic radiological event.

It is clear to me that for all of the above reasons Indian Point should be closed. I declare that the statements made in this declaration are true and correct to the best of my knowledge.

Executed this 7rd day of December, 2007 at Spring Valley, New York.

Marilyn Elie Marilyn Elie

State of New York)
)ss.:
County of Rockland)

On the 7th day of December, in the year 2007 before me, the undersigned, personally appeared
MARILYN ELIE, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

Notary Public

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

EXHIBIT B

Exhibit B

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 Rockland County Conservation Association, Inc.

The Rockland County Conservation Association, Inc. (RCCA) P.O. Box 213, Pomona, NY 10970 was founded in 1930. It is stated within our mission that we are dedicated to the conservation of our natural resources, promote sound land use, advocate clean air and water quality, develop proper drainage, support energy conservation and preservation of natural beauty.

The history of RCCA is one that spans decades with dedicated volunteers working to conserve our natural resources, which supports the lifeblood of our community's health and quality of life. In the 1940's it was, in part, the work of this organization that saved High Tor in Rockland County from further destruction, by raising funds toward the purchase the property that was dedicated to the Palisades Interstate Park. The insights of our early members proved to be immeasurable to the landscape that overlooks the expanse of the Hudson River.

The organization has been involved in conserving the natural beauty and resources in the Mid Hudson in actions with other groups, such as the opposition in the 1960's to the Consolidated Edison's proposal to build a hydroelectric plant on the significantly prominent Storm King Mountain. Indeed, we raised our voice in opposition to the construction of Indian Point 1 and continue to raise concerns for the exposures presented by the Indian Point nuclear power facility on the banks of the Hudson River across from Rockland County.

RCCA served on the Highlands Preservation Initiative Working Group that resulted in the passage of the Highlands Conservation Act 2004. We continue with the efforts on the Highlands Coalition to conserve this region of significance.

Our members continued concern for the threat to our natural resources is not in opposition to technology rather it is for the conservation of the essence of what defines sustainable life choices. To consider the ongoing leaks of strontium 90 and tritium, or faulty sirens merely as technical difficulties is to undercut the significance of these very real threats to our well-being. The NRC should not issue a new superceding license to the operator for another 20 years.

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

signed this day November 28, 2007

Dorice Madronero , president

Sworn to before me on November 28, 2007

Notary Public

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

Exhibit B

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.	)	**LicenseNo. DPR-64**
Indian Point Energy Center Unit 3	)	**Docket No. 50-286**
License Renewal Application	)	

DECLARATION OF Rockland County Conservation Association, Inc.

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The organization has been involved in conserving the natural beauty and resources in the Mid Hudson in actions with other groups, such as the opposition in the 1960's to the Consolidated Edison's proposal to build a hydroelectric plant on the significantly prominent Storm King Mountain. Indeed, we raised our voice in opposition to the construction of Indian Point 1 and continue to raise concerns for the exposures presented by the Indian Point nuclear power facility on the banks of the Hudson River across from Rockland County.

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Our members continued concern for the threat to our natural resources is not in opposition to technology rather it is for the conservation of the essence of what defines sustainable life choices. To consider the ongoing leaks of strontium 90 and tritium, or faulty sirens merely as technical difficulties is to undercut the significance of these very real threats to our well-being. The NRC should not issue a new superceding license to the operator for another 20 years.

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

signed this day November 28, 2007

Dorice Madronero , president

Sworn to before me on November 28, 2007

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

EXHIBIT C

Exhibit C

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.	)	**LicenseNo. DPR-64**
Indian Point Energy Center Unit 3	)	**Docket No. 50-286**
License Renewal Application	)	

DECLARATION OF SIERRA CLUB ATLANTIC CHAPTER

My name is Susan Lawrence. I am the Conservation Chair for the Atlantic Chapter of the Sierra Club. I represent the Atlantic Chapter and approximately 45,000 Sierra Club members in New York State. Thousands of Sierra Club members live and work within the affected area of the Indian Point Nuclear power plant.

On September 11 2001, the 9/11 hijackers flew directly over the plant. It was later learned that these terrorists had planned to attack Indian Point before they decided to attack the World Trade Center.

There are chronic problems and life threatening hazards caused by Indian Point. In 2005 leaks of tritium where discovered accidentally 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, and duration of the leaks, and identifying how to stop and remediate the leaks remains unknown. Since then other leaks also have been discovered by accident, such as the April 7th 2007 steam leak.

Our elected officials, Federal, State and Local, and thousands of Hudson Valley and New York State residents have called for Indian Point closure and for an Independent Safety Assessment prior to consideration for re-licensing. Entergy has been unable to properly install the required siren system.

Indian Point has been leaking Strontium 90, tritium and cesium into the groundwater and the Hudson River. The Hudson River is continuing to be polluted as a result of the inaction of the owners and regulators.

Indian Point does not have an adequate, workable or fixable evacuation plan

Indian Point should not be sited where it is currently located in the most densely populated region of the country and on an earthquake fault.

The NRC can not issue a new superceding license to the operator Entergy for another 20 years. Indian Point should be closed immediately and decommissioned.

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

Executed this 5th day of December 2007, at Albany, NY.

Susan Lawrence

Susan Lawrence

subscribed and sworn to before me
this 5 day of Dec, 2007

NOTARY PUBLIC

RICHARD J. SCHAEFER
Notary Public, State of New York
No. 01SC5065321
Qualified in Albany County
Commission Expires Sept. 03, 2010

EXHIBIT D

D Exhibit D

In the matter of
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.) **LicenseNo. DPR-64**
Indian Point Energy Center Unit 3) **Docket No. 50-286**
License Renewal Application)

and

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 MARK JACOBS

My name is Mark Jacobs. I live at 46 Highland Drive, Garrison, New York and am a member of WestCAN. WestCAN 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 my home in Garrison, about 5 miles from Indian Point, for over 10 years and in the Hudson Valley for my entire life (other than my time at college). I have spent and continue to spend significant time both on and by the Hudson River. I am a boater with my own canoe. I enjoy canoeing on the Hudson with my friends and family.

In addition, I am the Co-director of Longview School, located in Cortland Manor about 4.5 miles from Indian Point. I often bring students down to the Peekskill Riverfront Park and Croton Point Park. During these visits, we often walk by and wade in the Hudson River.

I am very concerned about Entergy's petition to relicense Indian Point. The ongoing multiple leaks with undetermined sources and failed remediation are unacceptable. The Hudson River is a unique and vital resource to our community and the entire New York region. 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 and my students from enjoying the river.

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.
11/26/07
Executed this 26 day of November, 2007, at ~~Cortland Manor~~ Peekskill S.S., NY.

On the 26 day of November, in the year 2007 before me, the undersigned, personally appeared

Mark R Jacobs, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

Notary Public

Jennifer L. Saleski
Notary Public, State of New York
No. 01SA6077075
Qualified in Dutchess County
Commission Expires July 01, 2010

EXHIBIT E

f Exhibit E

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**License No.**
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.	)	**DPR-26+64**
Indian Point Energy Center Unit 2 + UNIT 3	)	**Docket**
		No. 50-247
License Renewal Application	)	No. 50-286

DECLARATION OF GARY T. SHAW

My name is Gary T. Shaw. I live at 9 Van Cortlandt Place, Croton on Hudson, NY 10520, about 6 miles from Indian Point, which is within the 10-mile radius of the Emergency Planning Zone, also known as the peak fatality zone. I am a member of the Westchester Citizens' Awareness Network (WestCAN), Hudson River Sloop Clearwater ("Clearwater"), Croton Close Indian Point and the Steering Committee of the Indian Pont Safe Energy Coalition (IPSEC).

WestCAN 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, Entergy Nuclear Indian Point 3, LLC and Entergy Nuclear Operations, Inc., License Renewal Application.

My family and 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 River is a unique and vital resource to our community and the entire New York region. Today, Indian Point could not be sited in Buchanan, NY due to 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 judged 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 river bank. Among the materials I have personally extracted are construction materials such as panels of house siding and aluminum window frames, car parts, tires and household appliances such as an 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 under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed this 3 day of December, 2007, at New York, NY.

Gary Shaw

Your name here

State of New York)
)ss.:
County of NY)

On the 3RD day of DEC, in the year 2007 before me, the undersigned, personally appeared GARY SHAW, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

Notary Public

PHILIPPE HANNA
Notary Public, State of New York
No. 01HA6100743
Qualified in New York
Commission Expires 4/27/11

EXHIBIT F

F Exhibit F

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C.	)	**License No.**
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.	)	**DPR-26+64**
Indian Point Energy Center Unit 2 + UNIT 3	)	**Docket**
		No. 50-247
License Renewal Application	)	No. 50-286

DECLARATION OF JEANNE D. SHAW

My name is Jeanne D. Shaw; I live at 9 Van Cortlandt Place, Croton on Hudson, NY 10520, about 6 miles from Indian Point, which is within the 10-mile radius of the Emergency Planning Zone, also known as the peak fatality zone. I am a member of the Hudson River Sloop Clearwater ("Clearwater"), Croton Close Indian Point, Westchester Citizens Awareness Network (WestCAN) and the WESPAC Foundation.

WestCAN 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, Entergy Nuclear Indian Point 3, LLC and Entergy Nuclear Operations, Inc., License Renewal Application.

I have lived in the Hudson Valley for more than 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 more, faster or larger leaks, will affect my ability 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 conditions 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 under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed this 4 day of December 2007, at Croton, NY.

Jeanne D Shaw

Your name here

State of New York)
)ss.:
County of Westchester

On the 4 day of December in the year 2007 before me, the undersigned, personally appeared Jeanne D Shaw, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

GLADYS G. OLIVA
STATE OF NY
NO. 01OL6040508
TERM EXPIRES
APRIL 24, 20 10
QUALIFIED IN DUTCHESS COUNTY
NOTARY PUBLIC

Gladys G Oliva

Notary Public

G Exhibit G

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C., and	)	**License No. DPR-26**
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.	)	**License No. DPR-64**
Indian Point Energy Center Unit 2 and	)	**Docket No. 50-247**
Indian Point Energy Center Unit 3	)	**Docket No. 50-286**
License Renewal Application	)	

DECLARATION OF JUDY ALLEN

My name is Judy Allen; I live at 24 Seifert Lane, Putnam Valley, NY 10579, about 16 miles from Indian Point, which is within the 50-mile radius of the peak injury zone. I am a member of WestCAN, the Beacon Sloop Club and Clearwater.

WestCAN 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, Entergy Nuclear Indian Point 3, LLC and Entergy Nuclear Operations, Inc., License Renewal Application.

I have lived in the Hudson Valley for 27 years. I met my husband at the Beacon Sloop Club. He sailed me down the Hudson from Peekskill to Croton in his Hobie Cat when I was about 8 months pregnant. I have sailed on the Clearwater, the Woody Guthrie and the Sloop Club's "chicken sloop" with him. I helped to get the River Pool at Beacon started; when the organization needed funding to build a prototype, I wrote the grant to DEC and we got $75,000 to construct it. We also received $10,000 from the Hudson River Foundation. My husband and my son have both participated in the annual fundraising swim from Newburgh to Beacon for the River Pool. I have eaten shad and shad roe at the annual shad festivals from Croton to Beacon to Newburgh. And I have written a play that takes place at the Bear Mt. overlook on Rt. 6 just south of the Bear

Mt. Bridge that was a finalist in a recent play festival in Garrison, NY. All this by way of saying my life is bound to and inspired by the Hudson River.

The Indian Point nuclear plants should never have been built. The facility sits directly on the Ramapo fault line. Where there is a fault, there will, sooner or later, be an earthquake. No matter how good the security, no matter how adept the control room operators, they cannot control the strength of an earthquake and we are all potential victims of an earthquake that may have terrible consequences.

Even before a terrible incident that could destroy the plant and the communities around it, the everyday detrimental effects of aging endanger our communities on an ongoing basis. Adverse aging effects resulting from metal fatigue, erosion, corrosion and shrinkage could affect a number of reactor and auxiliary systems - MANY individual components and structures. Overstressed equipment, reduced safety margins, and the possible loss of required plant functions (including Entergy's capability to prevent or mitigate the consequences of accidents with a potential for offsite exposures) are unacceptable.

How is Entergy going to effectively manage Indian Point 2 and 3's aging during the proposed period of extended operation? Has Entergy identified any additional actions, i.e. maintenance, replacement of parts, etc., that it will take to manage adequately the detrimental effects of aging? Have they indicated there's enough money available for this? Because apparently, it has recently been discovered that there ISN'T enough money for "refurbishment". Is that the same thing?

And forget about 20 more years; even without a tragic incident, we are victims of Indian Point's everyday operation. The planned releases of radioactive steam and water were

calculated "within acceptable limits" long before the National Academy of Sciences stated that there is NO safe level of exposure to radiation. Why weren't the "acceptable" limits changed to reflect this? And the unplanned releases in the form of leaks of strontium-90 and tritium are even worse. At present it is not known where the leaks originate, how long they have been going on, or their extent. However we DO know that they are reaching the Hudson River. All levels of river life are being exposed to radioactive elements. There is no evidence Entergy is taking this seriously; they are just plowing ahead with their application to re-license the plants for another twenty years. Remediation of the leaks MUST be tied to re-licensing and if they are not stopped, there's no re- or superseding license.

At the recent environmental scoping public meeting many workers at Indian Point said over and over again how safe the plant is and how they have no fear of working there. I believe them. I believe it is MUCH safer to work INSIDE Indian Point than to live nearby, because the workers are not subject to the radioactive steam and water emanating from the plant on a regular basis. According to the New York State Cancer Registry (http://www.health.state.ny.us/statistics/cancer/registry/zipcode/lung/westchester.htm), lung cancer cases of women in Buchanan, NY between 1999 and 2003 are 50-100% higher than expected, as they are in Lake Peekskill and Mohegan Lake. In Montrose, women's lung cancer rates are more than 100% higher than expected. Female breast cancer from the same source (http://www.health.state.ny.us/statistics/cancer/registry/zipcode/breast/westchester.htm) shows similar results.

Ignoring the environmental and health effects of Indian Point's current operation in order to re-license the plants for another 20 years is at best irresponsible and, more

likely, criminal. To do so would be to endanger the health, lives and well-being of 20 million people in the 50-mile radius of Indian Point.

Thank you.

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

Executed this 30th day of November, 2007, at Carmel, NY.

Your name here

State of New York)

)ss.:

County of Putnam)

On the 30 day of NOV, in the year 2007 before me, the undersigned, personally appeared

JUDY ALLEN, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

Notary Public

JEAN O'HARA
Notary Public, State of New York
No.4962680
Qualified in Putnam County
Commission Expires February 26, 2010

EXHIBIT H

H Exhibit H

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of		
ENTERGY NUCLEAR INDIAN POINT 2, L.L.C., and	)	**License No. DPR-26**
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C.	)	**License No. DPR-64**
Indian Point Energy Center Unit 2 and	)	**Docket No. 50-247**
Indian Point Energy Center Unit 3	)	**Docket No. 50-286**
License Renewal Application	)	

DECLARATION OF ELIZABETH C. SEGAL

My name is Elizabeth C. Segal; I live at 33 Fairview Avenue, Tarrytown, NY 10591, about 13 miles from Indian Point, which is within the 50-mile radius of the peak injury zone.

WestCAN 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, Entergy Nuclear Indian Point 3, LLC and Entergy Nuclear Operations, Inc., License Renewal Application.

I have lived in the Hudson Valley for my entire life – 52 years – and am very connected to the Hudson River. Over the past three years I have participated in seven Hudson River swims (six of them crossing the Hudson) to raise funds for the National MS Society, the National Leukemia and Lymphoma Society, and River Pool at Beacon. It is exhilarating and restorative to swim across the majestic Hudson, and I am profoundly grateful to be able to do so. I am also fortunate enough to have a fine view of the river from my living room window, and love watching it every day.

Elizabeth C Segal, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

Notary Public

SARAH CAMACHO
Notary Public - State of New York
ID No. 01CA6145045
Qualified in Westchester County
My Commission Expires May 1, 2010

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 Licenses 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

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	Docket No. 50-247
Consolidated Edison Company of New York, Inc.	)	
	)	

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 30, 1969

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.

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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 PROPOSED RULE PR-50
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.

C1/3

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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

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, [illegible] covered by Criterion 20

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.

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.

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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.

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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 5 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 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

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

TABLE OF CONTENTS

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

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.

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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 D. 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

1

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

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.

12

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

Supplement to
Exhibit
~~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

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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. Observe 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 procedure. 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.

EXHIBIT L

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	**- NOTES -** 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	**- NOTE -** Not required to be performed until 12 hours after establishment of steady state operation. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	72 hours

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level or discharge flow) monitor,

b. One containment atmosphere radioactivity monitor (gaseous or particulate), and

c. One containment fan cooler unit (FCU) condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1	- NOTE - Not required until 12 hours after establishment of steady state operation.	
		Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	A.2	Restore required containment sump monitor to OPERABLE status.	30 days

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level or discharge flow) monitor,

b. One containment atmosphere radioactivity monitor (gaseous or particulate), and

c. One containment fan cooler unit (FCU) condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1	- NOTE - Not required until 12 hours after establishment of steady state operation.	
		Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	A.2	Restore required containment sump monitor to OPERABLE status.	30 days

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EXHIBIT M

EN-28

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

RELEASED TO THE PDR
9/30/92
date initials

September 18, 1992

SML 10/12-16
#76

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-92-223 - RESOLUTION OF DEVIATIONS IDENTIFIED DURING THE SYSTEMATIC EVALUATION PROGRAM

The Commission (with all Commissioners agreeing) has approved the staff proposal in Option 1 of this paper in which the staff will not apply the General Design Criteria (GDC) to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. While compliance with the intent of the GDC is important, each plant licensed before the GDC were formally adopted was evaluated on a plant specific basis, determined to be safe, and licensed by the Commission. Furthermore, current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC. Backfitting the GDC would provide little or no safety benefit while requiring an extensive commitment of resources. Plants with construction permits issued prior to May 21, 1971 do not need exemptions from the GDC.

The Systematic Evaluation Program (SEP) should be closed. The staff should, however, continue the generic review of the SEP lessons learned and prioritize the issues in the Generic Safety Issue program.

cc: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque
OGC
OIG

92 SEP -5 AM 10:57

SECY NOTE: THIS SRM, SECY-92-223, AND THE VOTE SHEETS OF ALL COMMISSIONERS WILL BE MADE PUBLICLY AVAILABLE 10 WORKING DAYS FROM THE DATE OF THIS SRM

EXHIBIT N

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".

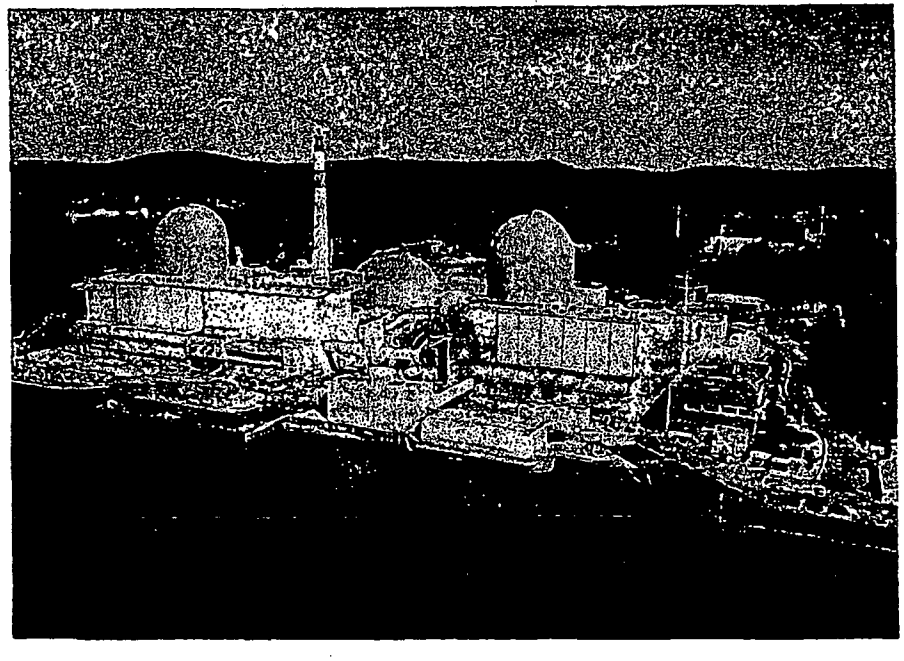
CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEM COMPONENTS (Category A)

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.

EXHIBIT P

Indian Point Units License Renewal

Indian Point Unit 1, 2 and 3



Presentation by Karl Jacobs

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Background

- Local Resident of Cortlandt Manor for 18 years
- Have never been in the employ of Entergy
- 20 years of experience with nuclear operations, maintenance, project management, installation of major multi million dollar safety related nuclear and non- nuclear equipment at IP3 that meets the required federal, state and industry accepted codes.
- 20 years of experience primarily on Indian Point Unit 3 in developing and implementing aging management programs for the Reactor Vessel, Reactor Internals, Pressurizer, Rector Coolant Piping and Steam Generators etc.
- Participated in the License Renewal rulemaking (10CFR50.54a) as IP3 Utility representative and as a Westinghouse Owners Group (PWR NSSS) Subcommittee Chairman, Nuclear Energy Institute (NEI), Electric Power Research Institute and the Nuclear Regulatory Commission
- Lead Technical Engineer for the technical and economical studies for Indian Point Unit 3 and James A. Fitzpatrick Nuclear Plant License Renewal evaluations. The IP3 studies were performed for the previous owner are identified.
  - License Renewal Comparison of IP3 design, operation and performance characteristics to the Industry Pilot Plant (Surry 1) .
  - Life Extension/ License Renewal Program Technical Summary Report
  - Cost/Benefit Analysis

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Highlights of the 10CFR 50.54 and revised 10CFR51 Rule

Identification of the License Renewal Components for scoping and screening evaluations and if determined technically that a component does not meet the additional life extension requirements (an aging management programs would be identified for implementation (on –going current licensing basis programs, newly developed and required to be implemented during their license renewal period)

- This scoping is also to include the identification and evaluation of time limited aging analysis (TLAA)

Environmental Impact Studies – Opens the door for Cooling Towers to be evaluated and possibly installed in lieu of present Water Cooled Condenser System – The Cooling Towers would help address the zebra mussel issues which are an environmental issue that in the past has plagued the safety related service component and service water cooling systems for IP3 and IP2. (Reduction and possible removal of their chlorination injection program, will also benefit the Hudson River.)

Identify and /or develop aging management programs of the components that are identified through the screening process for managing aging effects and address TLAAs

Emergency Planning and Security is not part of the 10CFR50.54 and revised 10CFR50.51 rule and needs not to be addressed under License Renewal Application

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Indian Point Unit 1 License Renewal Scoping Issues

The license renewal application (LRA) is for IP2, IP3 and shared systems with IP1

A review of the scoping of components in the LRA the does not identify Indian Point Unit 1 Containment structure and spent fuel systems and their support systems as being part of the License Renewal Application. See LRA Section 2.4.1 Describes only Unit 2 and Unit 3 Vapor Containment Structure. Unit 1containment structure is omitted.

Per the License Renewal Application for IP2 and IP3 under containment scoping and screening review in section 2.4.1 page 2.4.-2 state "the containment buildings have the following intended functions for 10CFR54.4(a)(1), (a)(2) and (a)(3)."

- **Provide support, shelter and protection for safety- related equipment**
- **Maintain essential leak tight barrier**
- **Maintain integrity such that safety –related equipment is not affected.**

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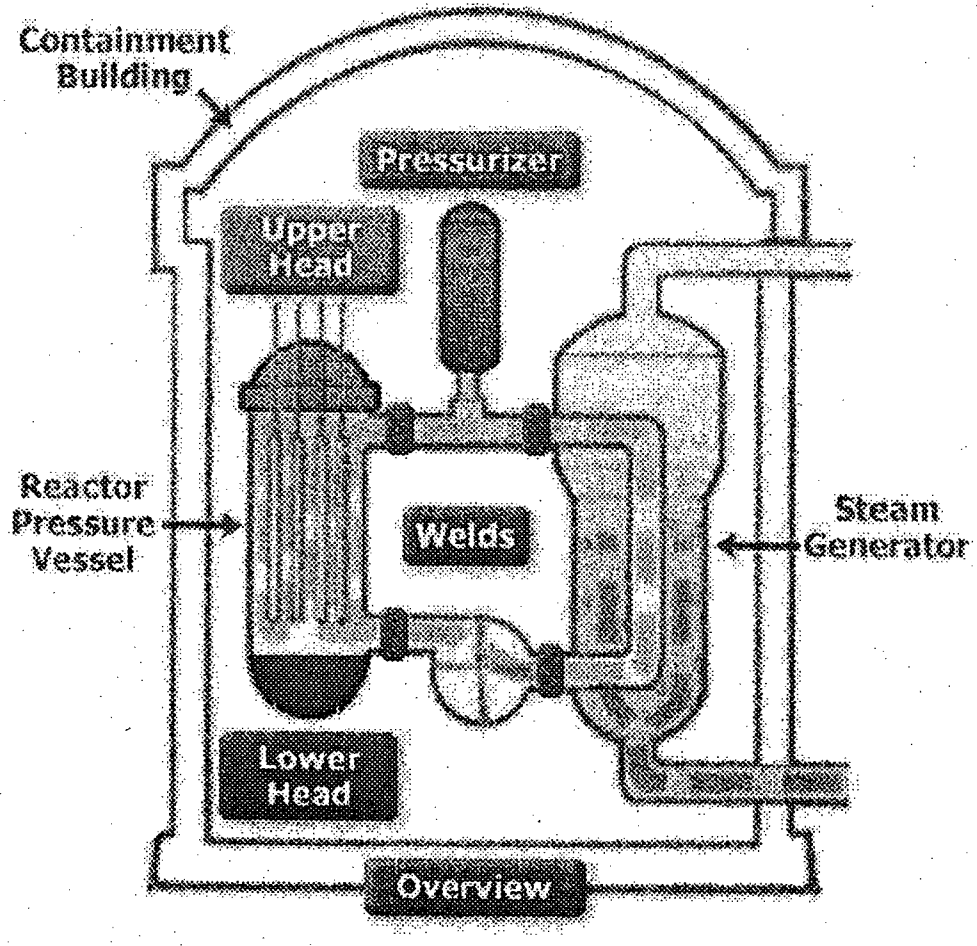
Indian Point Unit 1 License Renewal Scoping Issues

Indian Point Unit 1 supports the spent fuel cooling system is located in the containment structure. IP Unit 1's containment performs intended functions as defined by the License Renewal rule function above. In addition other scoping of license renewal scoping and screening systems are inside the containment structure that have been excluded are spent fuel pools structures; HVAC filtration for radioactive airborne particulates, containment penetrations, spent fuel pool system cooling piping and their supports, spent fuel cooling pumps, instrumentation for monitoring the operations of the spent fuel system, electrical wiring, spent fuel bridge,cranes and radiation monitors etc.

With the Entergy IPEC LRA allowing for IP Unit 1 shared components to be included in their application has opened a doorway to allow for a full scoping and screening of IP Unit 1 systems and components to protect the health and safety of the public

IP2 and IP3 Typical RCS Integrity Boundary



Reactor Vessel and Reactor Internals Typical to IP2 and IP3

Westinghouse Nuclear Steam Supply System Designer and Fabricator of Reactor Internals

Combustion Engineering is the Reactor Vessel Fabricator

IP2 RPV Construction Code – ASME Section III 1965 Edition

IP3 RPV Construction Code – ASME Section III Edition Winter 1965 Addenda

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Reactor Vessel (RPV)

- **Reactor Vessel Major Intended Functions**
  - Maintain the reactor pressure boundary
  - Support and contain the reactor core and core support structures
  - Support and guide reactor controls and instrumentation
  - Contain the reactor coolant around the reactor core and direct the coolant flow into the core and out into the reactor coolant piping and upper head
  - Interface with the RPV supports to provide a load path to the structural concrete
- **Subcomponents subject to an aging management review**
  - All of its subcomponents are passive, and only two of the subcomponents do not require an aging management.
  - There are only two subcomponents that do not require an aging management review. The RPV O-Rings, O-ring leak monitoring tubes and the refueling seal ledge do not support any RPV intended function

Reactor Vessel (RPV)

- For RPV neutron embrittlement is a critical aging management failure mechanism issue that must be accurately evaluated for License Renewal for both IP2 and IP3 reactor vessels.
- This IP3 reactor vessel has a projected RTndt value that would have exceeded the 10CFR50 Appendix G criteria during life extension if the criteria was not revised by the NRC
- For IP3 the lower shell plate (B2803-3) is the limiting RPV plate material.
- The projected RTpts for this same lower shell plate is very close to the 10CFR50 Appendix G criteria for the end of license renewal. With augmented aging management programs being implemented which are low leakage fuel management for neutron flux reduction,significant expansion of the reactor vessel surveillance capsule monitoring program, implement research and development programs on material crack initiation and crack growth with similar low fracture toughness' properties, along with a higher frequency of volumetric examinations of the RPV beltline than the present frequency requirements of ASME Section XI and Regulatory Guide 1.150 the RTpts may be successfully managed to meet life extension.
- For the same plate, the projected upper shelf fracture toughness energy for 60 calendar years is less than 10CFR50, Appendix G minimum criteria of 50- ft-lbs. This is a critical issue, that Entergy will need the NRC's assistance in a 10CFR50 Appendix G rule change to revise the criteria to a lower threshold value. This plate was originally installed with an initial +74 RTndt value. This was a fabricator miscue to allow the original installation of a shell plate in the Reactor Vessel Beltline with a +74 RTndt material property value to be installed. The plates that are installed in reactor vessels should have minimum initial Rtndt value of zero or a minus value to support Reactor Vessel longevity.

RPV

- The IP3 Reactor Vessel's lower upper shelf energy (a physical/mechanical properties of the RPV vessel wall) is a major concern for its lower shell plate B-2803-3. This plate material will not meet 10CFR50 Appendix G "Fracture Toughness Requirements" for license renewal. This plate is predicted to fall well below the 50 ft-lbs as measured by Reactor Vessel Surveillance Capsules charpy v –notch specimen testing.
- IP3 has two alternative approaches which are not even mentioned.
    1. An analysis is performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation. The margins against fracture must be equivalent to those required by the ASME Code, Section III, Appendix G
    2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests. The problem with this approach is the IP3 Reactor Surveillance Program remaining capsule specimens do not have the limiting plate material B2803-3 in any of this capsules. The statement made by Entergy in the license renewal application Section B.1.32,titled (Reactor Vessel Surveillance) page B-112 under the described enhancements that "The specimen capsule withdrawal schedules will be revised to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation."

RPV

- The IP Unit 3 RPV has an on going aging management program called reactor vessel surveillance capsule monitoring program that is in effect for its current licensing basis and does an extra capsules installed in the reactor vessel for life extension. The limiting plate material (B2803-3) as does not have any material in the remaining capsules to monitor the lower shell plate. This also will have a significant effect on their heat and cool down curves which are developed from the most limiting vessel plate material.
- The IP3 RPV materials has been volumetric examined (ultrasonic techniques) thoroughly every ten years from initial operation and no reportable indications were found. The volumetric equipment used for these inspections are very good, use a array of ultrasonic transducer probes, with high detection capabilities, sizing and locating any flaws are also very good. Please note that the RPV beltline is 100% inspected but access to allow for 100% of all RPV welds volume is not achievable do to interferences.
- Bottom line IP3 reactor vessel beltline plate material absolutely does not support license renewal unless the NRC revises 10CFR50 Appendix G requirement of maintaining a higher USE value of 50 ft –lbs for its belt line material

RPV

Present Industry Events and experiences has identified that the IP2 and IP3 Reactor Vessels' Heads must be replaced prior to life extension. This is a generic industry concern for the Westinghouse Reactor Vessel Heads' penetration tube welds that started in September 1991 @ the Bugey Unit 3 PWR nuclear plant in France. Then in May 1992 Ringhals Unit 2 , a Westinghouse- designed PWR –in Sweden found a 25 % around through wall crack in the CRDM penetration. Then it came to America. 1995 DC Cook Unit 2 (Westinghouse design) a crack measured as the deepest point of 6.88mm, 25% around the CRDM tube wall. VC Summer Plant was next, then Ringhals 3 and 4in June 2001, then Oconee and an Entergy Plant ANO-1. NRC Bulletins have been issued.

- NRC Bulletins 2001 –01 Circumferential Cracking of Reactor Pressure Head Penetration Nozzles
- NRC Bulletin 2002-01 Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity
- NRC Bulletin 2002-02 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs

- Entergy LRA response – Intend to use ASME Section XI, Sub Section IWB Inservice Inspection and Water Chemistry Control Programs. Detection of Cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination (underside of the head) techniques.
- Entergy has not realized as a company that safety and lowering the risk to public health comes first not economics This is real cracking issue that many same design plants are experiencing now! This cracking can lead to a control rod missile ejection followed with a small break loca. This failure would permanently shut IPEC down!
- Reactor Coolant Supports are located in a difficult to access area and limits inspection capabilities. Reactor Coolant Supports can corrode since the are serviced with cooling water. A inspection program to fully assess these reactor supports and cooling system requires a definitive aging management program.

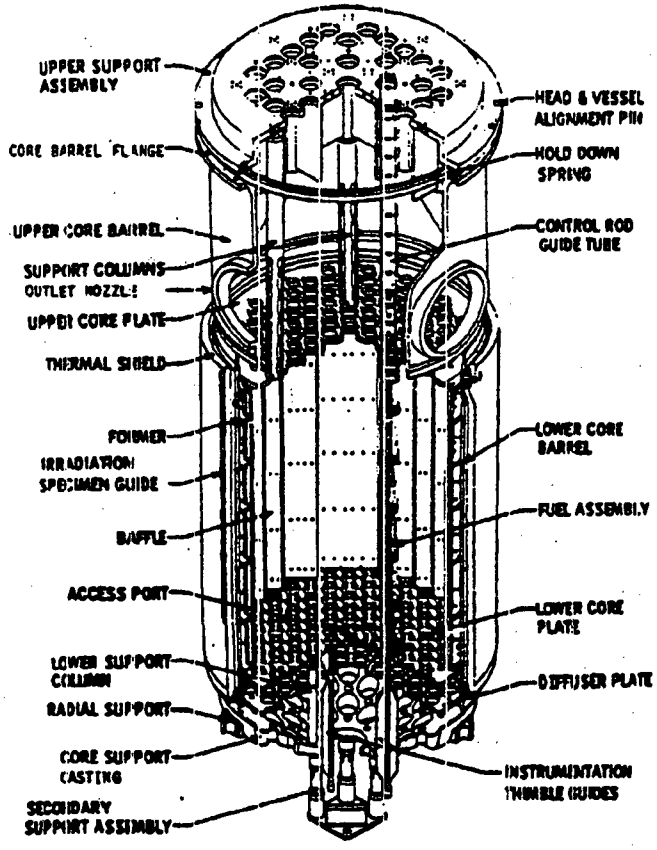
PZR

- Aging Degradation mechanisms for the IP 2 and IP3 Pressurizers are so significant that replacement is the only option for License Renewal. Some Highlights
- The pressure boundary materials of the Pressurizer are susceptible to Primary Water Stress Corrosion Cracking (PWSCC)
  - The pressurizer has Inconel 82/182 weld metal in pressurizer safety, relief, spray and surge nozzles which is susceptible cracking due to PWSCC
- The pressure boundary materials of the Pressurizer have significant end of life fatigue issues that will not meet life extension time frame
  - Fatigue of the upper portion of the pressurizer shell (44 years), the spray nozzle(49 years), the manway bolts (46 years), the seismic support lugs(41 years), lower head (due to insurge/outsurge transients), the heater wells (due to insurge and outsurge transients), the surge nozzle, the support skirt and flange (skirt –to-lower-head weld 54 years).
  - Then when you impose the NRC environmental effect to the fatigue calculations the list gets longer. Lower head (42 years), the safety and relief nozzles (53 years) and instrument nozzles (51 years)
  - This is back up by the NRC Final Safety Evaluation Report on the Acceptance for Referencing of a Generic License Renewal Program Topical Report by the Westinghouse NSSS Vendor "License Renewal Evaluation: Aging Management Evaluation For Pressurizers" dated October 26, 2000
  - Aging Management Program 2.3 needs to be imposed. This states that if the TLAA can not show acceptable usage for the license renewal period, the fatigue adequacy will be met by implementing a repair and replacement program in accordance with ASDE Section XI IWA- 004000 or IWA-7000
- NRC has issued a Final safety Evaluation Report for "Acceptance for referencing of Generic License Renewal Program Topical report entitle, "License Renewal Evaluation Aging Management Evaluation for Pressurizers" WCAP-14574 Revision 0, July 1996

Reactor Vessel Internals

- Aging Management Evaluation for Reactor Internals – WCAP –14573
- WCAP –14573 was submitted to the NRC by the Westinghouse Owners Group for IP unit 2 and Unit 3 and received a NRC Safety Evaluation Report accepting this WCAP to support License Renewal
- Reactor Vessel Intended Functions
  - Ensuring the capability to shut down the reactor and maintain it in a safe shutdown condition
  - Providing (Non – Safety Related) intended Functions that support the function listed above
  - Ensuring the integrity of the reactor coolant pressure boundary (Bottom Mounted Instrumentation Flux Thimbles Only)

Reactor Vessel Internals



AGING MECHANISMS CONSIDERED

- IRRADIATION EMBRITTLEMENT
- STRESS CORROSION CRACKING
- IRRADIATION-ASSISTED STRESS CORROSION CRACKING
- EROSION and EROSION/CORROSION
- CREEP/IRRADIATION CREEP
- STRESS RELAXATION
- WEAR
- THERMAL AGING
- CORROSION
- FATIGUE
- SWELLING

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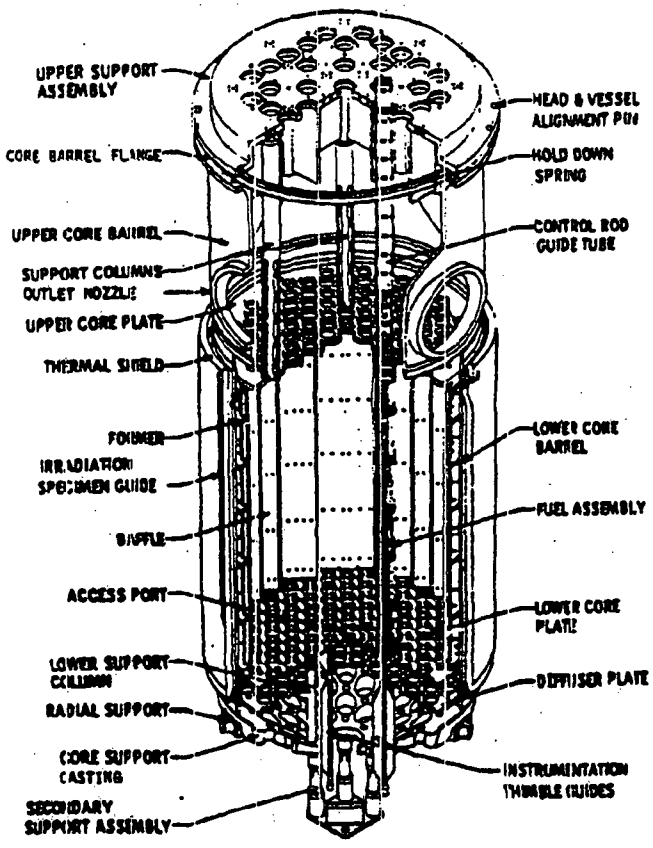
Reactor Vessel Internals



SUMMARY OF REACTOR INTERNALS SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

Part or Subcomponent	Aging Management Review Required?
Lower core plate and fuel alignment pins	YES
Lower support forging or casting	YES
Lower support columns	YES
Core barrel and core barrel flange.	YES
Radial support keys and clevis inserts	YES
Baffle and former plates	YES
Core barrel outlet nozzle	YES
Secondary core support	YES
Diffuser plate	YES
Upper support plate assembly	YES
Upper core plate and fuel alignment pin	YES
Upper support column	YES
Guide tube and flow downcomers	YES
Upper core plate alignment pin	YES
Holddown spring	YES
Head and vessel alignment pins	YES
Control rod	NO
Drive rod	YES
Neutron panels/thermal shield	YES
Irradiation specimen guide	YES
BMI columns and flux thimbles	YES
Head cooling spray nozzles	YES
Upper instrumentation column, conduit, and supports	YES
Mixing device	YES
Bolts and locking mechanisms	YES
Specimen plugs.	YES

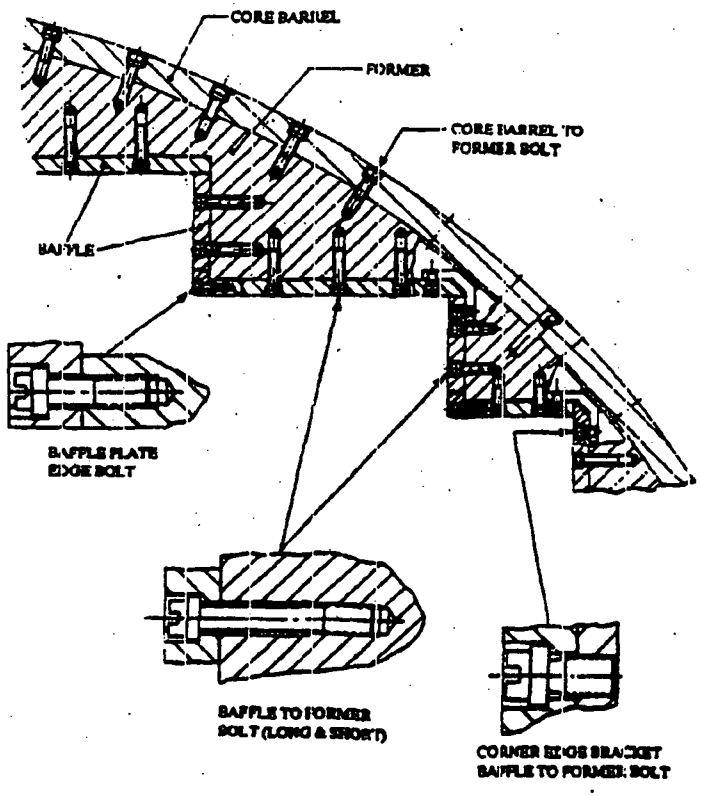
Reactor Vessel Internals

The following actions are needed for reactor vessel internals life extension as a minimum.

1. Control Rods Replacement for both Units 2 & 3
2. Specific fatigue monitoring programs for numerous Reactor Vessel Internals parts that are fatigue sensitive.
    1. Baffle Former Bolts
    2. Barrel Former Bolt
    3. Lower Core Plate
    4. Lower Support Plate
    5. Radial Key Weld
    6. Core Barrel Nozzle Weld
    7. Guide Tube/flow downcomers
    8. Upper support plate assembly

   Note these fatigue sensitive parts as calculated do not include the NRC request to include environmental effects.
3. Replacement Program for Baffle Former Bolts as a Lead Indicator for the other plant and for managing Barrel Former Bolts aging degradation. Cracked Baffle Bolts have already been replaced at Point Beach Unit 2 and RC Ginna Nuclear Power Plant in upstate New York.
4. Wear Management program for BMI flux Thimbles; Upper core plate alignment pins; radial keys and clevis inserts Per Commitments to NRC I&E Bulletin 88-09
5. Split Pin Replacement for Unit 2 with flexure modification to flexure less insert . with split pin replacement results from Unit 2, the results could be a lead indicator for Unit 3 aging management for split pins. This is only to be considered for mitigating the consequences of loose parts in the Reactor Vessel, Reactor Internals, and protection of the Steam Generators' tube sheet.

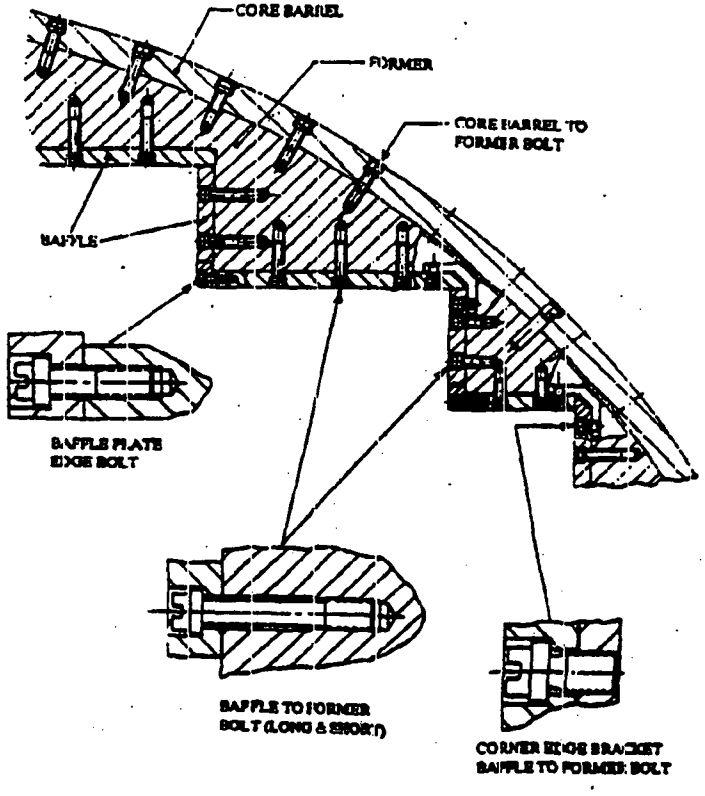
Reactor Vessel Internals



ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES FOR AGING MANAGEMENT OF BAFFLE/FORMER BOLTS (AMP-4.6)

Attribute	Description
Scope	Effects of cracking caused by fatigue, irradiation-induced changes in material properties, and irradiation-induced changes in stresses
Surveillance Techniques	• Visual inspection per Examination Category B-N-3 of ASME Section XI, Subsection IWB and Draft Subsection IWG • Loose parts detection monitoring system • Chemistry RC detection system • Augmented inspections (e.g., ultrasonic inspections)
Frequency	• Monitor with loose parts detection system • Monitor with RC chemistry detection system • ASME Section XI requirements, IWB-2410, -2411, -2412, -2420, -2430 and Draft IWG-2410, -2420, and -2430 • Perform sample baseline inspections prior to LR term with enhanced frequency in accordance with corrective actions
Acceptance Criteria	• Acceptable RC chemistry per technical specifications and • No loose parts from baffle/former bolt assembly and • Fatigue management program in Figure 4-1 and • Number of acceptable bolts and location ≥ the minimum number and location required to maintain core coolability and departure from nucleate boiling ratio (DNBR) within CLB limits, or if needed, for justification of continued operation (JCO), number of acceptable bolts and location ≥ JCO assumptions
Corrective Actions	The following courses of action depend on the bolt condition determined by the monitoring and inspection programs: • Supplemental examinations, analytical justifications or repair/replacement when relevant conditions are detected • Visual inspections, baffle gap measurements, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when baffle/former bolt assembly loose parts are detected • Fuel inspections, visual baffle plate inspections, baffle gap measurements, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when RC chemistry limits are violated • Adjustment of frequency of inspections and coverage • Analytis (e.g., fracture mechanics techniques, risk-based technology, advanced thermal/hydraulic methodologies) • Bolt replacement of a sample set so the existing bolts with indications may be analyzed (materials testing) and the new bolts monitored • Follow actions prescribed in fatigue management program
Confirmation	Acceptable performance per • Loose parts monitoring and RC chemistry programs • Augmented examinations (e.g., baffle gap inspections, ultrasonic examinations) • Analytical justification

Reactor Vessel Internals



ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES FOR AGING MANAGEMENT OF CORE BARREL/FORMER BOLTS (AMP-4.7)

Attribute	Description
Scope	Effects of cracking caused by fatigue, irradiation-induced changes in material properties, and irradiation-induced changes in stresses
Surveillance Techniques	• Visual inspection per Examination Category B-N-3 of ASME Section XI, Subsection IWB and Draft Subsection IWG • Loose parts detection monitoring system • Augmented inspections
Frequency	• Monitor with loose parts detection system • ASME Section XI requirements, IWB-2410, -2411, -2412, -2420, -2430 and Draft IWG-2410, -2420, and -2430 • Perform sample baseline inspections prior to LR term with enhanced frequency in accordance with corrective actions
Acceptance Criteria	• No loose parts from barrel/former bolt assembly and • Fatigue management program in Figure 4-1 and • Number of acceptable bolts and location ≥ the minimum number and location required to maintain core coolability and DNBR within CLB limits, or, if needed, for JCO, number of acceptable bolts and location ≥ than JCO assumptions.
Corrective Actions	The following courses of action depend on the bolt condition determined by the monitoring and inspection programs: • Supplemental examinations, analytical justifications or repair/replacement when relevant conditions are detected • Visual inspections, augmented inspections (e.g., ultrasonic inspections), analytical justifications or repair/replacement when barrel/former bolt assembly loose parts are detected • Adjustment of frequency of inspections and coverage • Analysis (e.g., fracture mechanics techniques, risk-based technology, advanced thermal/hydraulic methodologies) • Bolt replacement of a sample set so the existing bolts with indications may be analyzed (materials testing) and the new bolts monitored • Follow actions prescribed in fatigue management program
Confirmation	Acceptable performance per • Loose parts monitoring program • Augmented examinations (e.g., ultrasonic examinations) • Analytical justification

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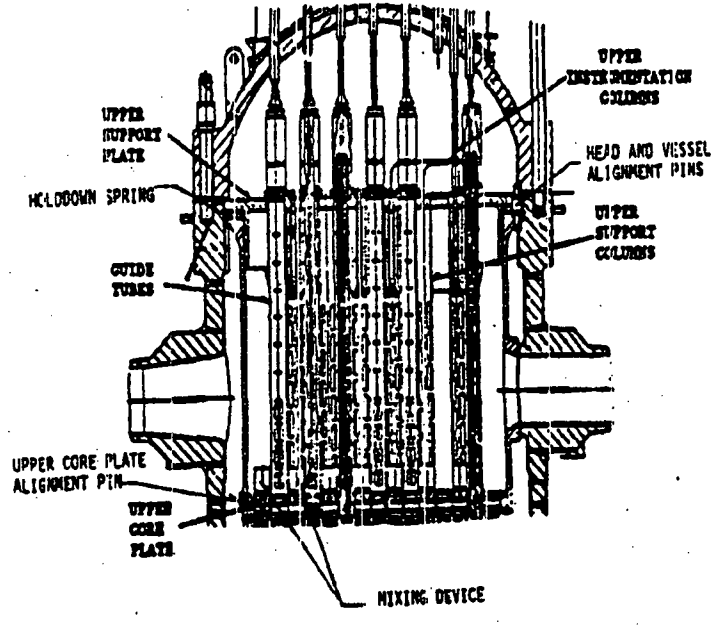
Reactor Vessel Internals



Upper Head Aging Parts that require Aging Management Efforts

- **Guide Tubes (Guide Tubes) – Wear**
- **Control Rods – Wear, Cracking -**
- **GT Flexure Replacement for IP Unit 2- Original Flexures are susceptible to SCC**
- **Split Pins – Stress Corrosion Cracking**

Exhibit Q1

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the matter of

ENTERGY NUCLEAR INDIAN POINT 2 L.L.C.,	)	
ENTERGY NUCLEAR INDIAN POINT 3, L.L.C,	)	License No. DPR 26 and
And Entergy Nuclear Operations, Inc.	)	License No. DPR 64
and Entergy Northeast, Inc.,	)	
regarding the Indian Point Energy Center	)	Docket No. 50-247 and
Unit 2 and Unit 3License Amendment	)	Docket No. 50-286
Regarding Fire Protection Program	)	

SECOND DECLARATION OF ULRICH WITTE
PETITION FOR LEAVE TO INTERVENE, REQUEST FOR HEARING, AND CONTENTIONS REGARDING LICENSE RENEWAL OF INDIAN POINT UNIT 3 AND UNIT 2
RE: CONTENTIONS 1~~3-17~~ 22-26

My name is Ulrich Witte. WestCAN, RCCA, PHASE, and the Sierra Club—Atlantic Chapter have retained me under the auspices of the Indian Point Safe Energy Coalition as a consultant with respect to the above-captioned proceeding. I am a mechanical engineer with over twenty-six year's professional experience in engineering, licensing, and regulatory compliance of nuclear commercial nuclear facilities. I have considerable experience and expertise in the areas of configuration management, engineering design change controls, and licensing basis reconstitution. I have authored or contributed to two EPRI documents in the areas

of finite element analysis, and engineering design control optimization programs. I have led industry guidelines endorsed by the American National Standards Institute regarding configuration management programs for domestic nuclear power plants. My 26 years of experience has generally focused on assisting nuclear plant owners in reestablishing fidelity of the licensing and design bases with the current plant design configuration, and with actual plant operations. In short, my expertise is in assisting problematic plants where the regulator found reason to require the owner to reestablish competence in safely operating the facility in accordance with regulatory requirements. My curriculum vitae is attached hereto as Attachment A.

I submit the following comments in support of each coalition stakeholder in Contentions 13-17 regarding the original design, construction and operation of the the plant, and their relevancy to the license renewal application as delineated in 10CFR Part 54.21, "Contents of the application,-general information" and 10CFR50.54.22 , "Contents of the application – technicial information," and 10CFT54.31 "Continuation of the CLB and conditions of renewed license" as contained in the License Renewal Proceedings of Indian Point Unit 2 and 3.

Contention:

The Applicant violated the Administrative Procedures Act in bypassing the Code of Federal Regulations (CFR) and instead used trade guidance for Indian Point 2 as opposed to 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

The design criteria based upon trade guidance was misrepresented by the Applicant in the renewal application as conforming to draft criteria published in 1967, and then relieved of all conformance to essentially all committed design criteria under a letter published by the Office of Nuclear Reactor Regulation in 1992.

The historical record shows that the applicant after discovering the error, failed to remediate the violation, and the misrepresentation, and therefore, is in violation of the Administrative Procedures Act.

This 40 year old design criteria problem affects both plants, and leaves Indian Point without adequate safety margins and the New York Metropolitan region without adequate assurance of protection of public health and safety

Submitted with particularity and specificity are provided here in for Unit 2. Unit 3 contains a similar historical record. The records show that the issue exists for both plants.

In essence, the aging management program required for license extension is predicated upon a sound, compliance and complete design basis record. With out this, the plant's material condition, basis design assumptions required for license renewal cannot be substantiated by prerequisite in situ conditions of essentially all aspects of each ageing plant. You cannot put a new roof on an old house constructed with meeting building codes, with numerous problems and is now falling , and then claim the house is now good for another 20 years.

With more than 26 years in licensing, design engineering control, configuration management. Establishing the legal ground for what Indian point 2

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received its original operating license against should be straight forward in particular given the 64 design criteria that provide the fundamental framework.

Essentially every other element of safely and hinges on respect for the licensing and design basis, and compliance with the law, and lawful operation of the facility. One would think one could simply examine the SER, along with the rest of the CLB circa the original operating license granted and find transparent the records for design basis, construction, licensing conditions, maintenance and safe operation of the plant.

After careful examination of the facts, as represented in the table of events, it appears that just the opposite is true. Applicable rules as found in 10 CFR are not followed, and in fact it appears the applicant and the regulator are doing the opposite routinely. Bypassing the core protection providing to the public under the Administrative Procedures Act.

The past and present owners of Indian Point have failed for forty years to ensure that the nuclear reactor(s) are in compliance with regulations established by the US Nuclear Regulatory Commission to ensure public health and safety.

In its application for a 20-year license extension, Entergy has misrepresented the official record of the Federal Register to give a false appearance of compliance with regulations. In fact, the reactor has been out of compliance since it was granted

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its original operating license 40 years ago.

The License Renewal Rule requires the applicant to identify which set of rules and regulations the reactor complies to (NRC regulations have been changed and updated several times since the 1960's.) However, the Applicant and the NRC are unable or unwilling to state which regulations are applicable to Indian Point.

The Nuclear Regulatory Commission has failed in its responsibilities by allowing Indian Point to operate under a set of "guidelines" proposed forty years ago by an industry lobbying group, but never approved by the NRC's mandatory "rule-making" process.

In its application for license extension, Entergy has failed to describe aging programs for the reactor's systems. Instead, they have "promised" compliance at some future date and time—after license renewal approval. This is a clear violation of the NRC's mandated procedures for license renewal. Indian Point 2 has serious environmental issues with spent fuel pool leaks, and radioactive leaks from underground piping that have not been addressed in their license renewal application.

The results of this are painfully obvious. A plant that that experienced a design basis event tube rupture, spent fuel pools leaking, and piping leaking. The most recent leak is cited under Contention 5, and was reported only a few weeks

Exhibit "Q1"

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

Executed this 8th day of December, 2007.

Ulrich K. Witte

State of New York)
)ss.:
County of Rockland)

On the 8 day of December, in the year 2007 before me, the undersigned, personally appeared Ulrich K. Witte, personally known to me or proved to me on the basis of satisfactory evidence to be the individual(s) whose name(s) is (are) subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their capacity(ies), and that by his/her their signatures(s) on the instrument, the individual(s) or the person upon behalf of which the individual(s) acted, executed the instrument.

SUSAN HILLARY SHAPIRO
Notary Public - State of New York
No. 02SH6060466
Qualified in Rockland County
My Commission Expires June 25, 2014

Notary Public

Attachment A

Ulrich K. Witte

71 Edgewood Way
Westville, Connecticut 06515
Home: 203 389 7374
Office: 860 577 8077
Mobile: 860 391 1183

Summary:

Over twenty-six years' of professional experience in engineering, configuration management, licensing, regulatory compliance of large scale commercial nuclear facilities. This includes management and implementation of design change control programs, engineering standards programs, multi-department/multi-functional licensing initiatives, plant design basis and engineering process improvement programs for six energy companies operating seven nuclear power plants. Responsibilities include:

- Systems solutions to plant operations, engineering modifications, safety analyses, design changes, installation and testing, software, drawing change programs, and training. Optimized function interfaces to insure proper coordination and synchronization for cost effective and compliant operation of the facility.
- Technical support management, and issue resolution programs that identified potential hardware, operational or equipment function issues, as well as document problems, data management problems and organizational enhancements
- Engineering Change Processes from change inception to document close-out
- Multi-department Configuration Management Program including technical approach, consensus, approval, and implementation. Managed a standing Configuration Management Programs Group whose goal was to integrate ten functional areas under a corporate strategic plan encompassing two nuclear facilities.
- Vertical slice system design/operation reviews, design bases / regulatory rule reconciliation, and licensing bases reconstitution and transitioning projects
- Integration of plant equipment information systems with business processes within engineering, materials management, maintenance, and plant operations.
- Structured business process modeling. Application of functional analysis purely from a data prospective—to enhance change management, efficiency.
- Chaired ANSI certified industry guidance on cost effective, compliant, and institutionalized programs for successful configuration management enhancement
- EPRI guidance on optimizing the Engineering Change Process
- Formal training to engineering department personal with specific courses on the engineering change process, plant safety analysis, and modification testing. Trained engineering personal on the requirements of the plant wide Configuration Management Program.

Technical Consultant

Northern Lights Engineering, L.L.C., 71 Edgewood Way, Westville, Connecticut 06515 (May 2002 –Today)

Established a consulting practice where I provided expertise in matters affecting the safe operation and regulatory compliance of commercial nuclear power facilities. This includes licensing andregulatory compliance issues, modification and implementation of industry standards, engineering design reviews, and configuration management analysis associated with an unexpected event, a design failure, or an elevated risk condition, and includes review of proposed changes to the plant operating license in preserving design efficacy.

Technical Advisor and Expert Witness to the law firm of Shems, Dunkiel, Kassel, & Saunders, PLLC
Currently providing technical assistance in prefiled testimony regarding Entergy Nuclear Operations application for renewing the operating license of Vermont Yankee. This includes Aging Review Program, in particular flow-accelerated corrosion issues, and finite element fatigue analysis reviews of susceptible components and a number of other contentions related to the safe operation of the plant beyond its 40 year license at 120% of originally design power.

Technical Advisor, to the law firm of Leroche, Meyers, and Conswel, LLP.
Provided licensing and regulatory compliance expertise in legal claim and derivative action by the board of directors of the First Energy Corporation against its corporate officers in their role associated with the Northeast black out of August 2003, and the mismanagement of the Davis Besse Nuclear Power Plant.

Technical Advisor to the Union of Concerned Scientists
Provided technical review of UCS analysis of the Davis Besse reactor head corrosion event. This included analysis of the loss of integrity of the reactor vessel, and the immediacy of the reactor head failure.

Senior Scientist, Dominion Resources Inc, Millstone Station:

P.O. Box 128, Waterford, Connecticut 06385-0128 (December 1996 – 2002)

Project Manager, Licensing Commitments. Established the Regulatory Commitment Management Program. Developed a program that established senior management and department level control of more than 30,000 licensing commitment that was previously broken. The substantially enhanced program captured, dispositioned, consolidated, and managed implementation of docketed commitments to the NRC. Status, responsibility and clear communication were successfully implemented to allow Millstone to successfully restart Units 2 and 3.

The effort required substantial procedure revisions, customer consensus building, and integration of separate free-standing department specific database applications, as well as the station wide action item tracking system. A near term deliverable necessary for the successful restart of Unit 3 was to provide a workable, compliant and functioning regulatory commitment management program.

Project Manager, 50.54(f) Licensing Bases Transition Project. I led a team of 14 individuals to disposition and validate approximately 5100 regulatory commitments necessary for restart of Unit 3. The effort has led to a quality rate of more than 98 percent with production average of about four hours per commitment.

Manager, Configuration Management Program, New York Power Authority:

123 Main Street, White Plains New York 10621, Nuclear Generation Department, Engineering Division
(November 1991 - November 1996)

Established the Configuration Management Program for the New York Power Authority's nuclear facilities. Included are 10 functional areas and integrated controls as authored in the corporate strategic plan. Management functions and technical skills include the following:

- Established Configuration Programs Group. This group and my position were established as a result of INPO Plant Evaluation calling for configuration management enhancement, and resolution of design control issues identified by the NRC in their DET Inspection of 1991 of the FitzPatrick Plant, as well as independent assessments. Recruited permanent staff, and supplemented the group with contracted staff on as needed basis to support both plants.

- Modified the engineering change process. Areas of immediate attention included the Design Control and Modification Programs, where a series of working groups were established to correct technical content and improve quality, ownership, and business efficiency of the design change process. This effort was achieved via: (1) a formal process to assess, model, and enhance the design change and modification process and interfaces to key functions; and (2) immediate changes to engineering procedures.

- Assessed and enhanced the Plant Equipment Data Base and controls for each plant. Results of the assessment indicated that the IP3 Plant Equipment Database contained significant problems with component classification, equipment type and status, maintenance history etc. Prepared and implemented a recovery plan and project team to reestablish the controls and content of database to be compliant with NRC Generic Letter 83-28 and to support the plant restart. Streamlined and enhanced the component classification process for both plants. Established controlled and non-controlled segregation of plant equipment in accordance with recent EPRI guidance.

- Automated and validated existing fragmented and corrupt sources of engineering information. These data sources were compiled, validated, and controlled and included multi-department areas such as set point controls, Electrical Cable and Raceway Information Systems for JAF and IP3, along with the fuse controls and data management.

- Developed design basis problem resolution process, "Design Document Open Item". Established methods for prioritizing, tracking and closing out design document issues. Established proper interface and control room notifications as per tech spec requirements. Provided guidance on operability determinations and reportability. Provided oversight for classifying and tracking more than 1100 open design issues for IP3 and 300 for JAF. Defended program to the NRC.

- Established working groups between Nuclear Generation Department and the corporate wide Information Management Organization. Gained management endorsement for areas of data quality improvement and automation for the Nuclear Generation Department. This led to enhanced implementation of the equipment information systems for both sites.

Project Manager, Program to Assure Completion and Quality, Tennessee Valley Authority:

(December 1990 - March 1991) Under contract by CYGNA Energy Services to the Vice-President, Engineering and Operations Department, Watts Bar Nuclear Plant.

- Developed a comprehensive plan to measure progress and confirm quality of the in-progress design evolution of the plant. Developed a methodology for linking specific plant equipment to that equipment's respective design basis (and associated design attributes); license commitments; and numerous verification programs currently in place. The five phase program was presented to NRR in January and received approval as an activity to assist TVA in removing the stop work order on construction of the facility.

Technical Manager, Configuration Management Program, Southern Nuclear Operating Company:

(December 1988 - November 1991). Under contract by ABB Impell and CYGNA Energy Services to Corporate Engineering Manager, Edwin I. Hatch Nuclear Plant, Georgia Power Company.

- Established and implemented the Hatch Configuration Management Program. Phase one of the effort included definition, establishment of management objectives, specification of the configuration management program scope and development of a reference manual.
- Developed and executed formal rigorous horizontal evaluations (the second phase of the project) of each relevant functional area including engineering design, implementation, plant operations and maintenance, procurement, information systems, document control and others. The program integrates functional areas across the plant, each architect engineer, and corporate (SONOPCO and Southern Company Services) organizations.
- Implemented enhancements to the program. This phase includes upgrading the design change process to achieve successful integration across organizations; stricter adherence to closure activities; and formal design engineering involvement in such activities as procurement of replacement items (equivalency). Additional controls were established such that misapplication of information obtained through informal design change processes such as the "Request for Engineering Assistance".
- Reconciling the plant's design basis. A second major activity of the program was to compile, consolidate, and ultimately, automate the plant's design basis. A major objective is to provide access and retrievability of current design basis to each of the key users of each participant organization.
- Applied Structured Business Analysis including CASE tools in the evaluation and enhancement phases. The as-found configuration management activities of all relevant processes were modeled and analyzed with this technique. Proposed enhancements are then tested on the model prior to actual implementation.
- Chaired the subcommittee for the Nuclear Information and Records Management Association which is developing a Technical Position Paper entitled, "Implementation of a Configuration Management Enhancement Program for a Nuclear Facility".

Team Leader, NRC Safety System Functional Inspection Response Organizations:

Led the NRC Safety System Functional Inspection Response Teams for Georgia Power Company (1989), and Sacramento Municipal Utility District (1987). Assisted as team coordinator in the GPC - Plant Hatch Electrical Distribution System Functional Inspection Response Team (1991). Under contract by ABB Impell (December 1987 - November 1990) to the site Engineering Manager, Rancho Seco, SMUD. and CYGNA Energy Services (December 1990 - November 1991) to the Corporate Engineering Manager, Edwin I. Hatch Nuclear Plant, Georgia Power Company.

- In the case of GPC, the NRC SSFI resulted in validation of the in progress implementation of the Hatch Configuration Management Program, and only one violation to the licensee.
- The effort included an SSFI self-assessment as well as managing the utility through the NRC inspection.
- For SMUD, developed and executed a plan for closure of both immediate findings and long term corrective action required. Assisted in defending the plan to the NRC.
- For GPC - Plant Hatch EDSFI in June 1991. Developed and implemented an EDSFI Preparation Plan for the Engineering (both A/Es) and site organizations. This effort included management of a 27 man team preparation and inspection response team for the Hatch EDSFI.

Deputy Mechanical Engineering Manager, Engineering Department

Under Contract to the Site Engineering Manager, Rancho Seco, Sacramento Municipal Utilities District, Rancho Seco (April 1986 - September 1987)

Managed the implementation and closure of over 400 modifications to the plant. Provided the NRC with a basis for allowing a successful restart of the facility. (January 1986 to November 1986) Impell Lead Project Engineer, Class 1 Piping and Support Recertification Effort, SMUD.

- Developed an engineering department action plan to improve technical quality, reconstitute design basis for five systems, control costs of plant modifications, and improve adherence to schedule.
- Responsible for the complete recertification of the Pressurizer Relief Line, Decay Heat System, and others. Responsible for expediting and implementing design changes as necessary through to closure. Assisted in Utility responses to NUREG-0737, and I&E 79-14.
- Upgraded the Engineering Department procedures to gain credit for the relaxation of ASME code requirements in structural damping values. Initiated the FSAR changes as well.

Project Engineer, Fire Protection:

Under Contract to Sacramento Municipal Utilities District, Rancho Seco (November 1984 to April 1986), SMUD Fire Protection Coordinator, Fire Protection Program

- Developed the SMUD Appendix R Fire Protection Program. Established or substantially revised 110 plant and engineering procedures including shutdown procedures on total loss of the plant's control room, technical specification surveillance procedures, fire protection system maintenance procedures, and the development of a fire protection program manual.

Successfully defended the program to the NRC during the 1985 Appendix R Inspection, with no resulting findings or open items.

Additional Experience (6/78 through 8/84):

Senior Engineer, performed original pipe stress analysis and support placement for Duke Power's Catawba Plant. Qualified approximately 8 class one and two plant systems. (ABB Impell 6/78 - 12/79).

Non-linear finite element analysis of large diameter piping for EPRI. Analysis of production stress codes versus non-linear evaluation techniques, versus actual in situ testing of the system. Results were published in EPRI Report "Seismic Piping Test and Analysis. (ABB Impell, 1980 -1981)

As Project Engineer, directed the preparation of the annual Emergency Plan exercises for Kansas Gas and Electric Company, Union Electric Company, and Texas Utilities. In two plants, the exercise was installed on the plants simulator, and received recognition from the NRC for realism of the scenario. (ABB Impell 1982-1984).

EMPLOYER SUMMARY:

Northern Lights Engineering, L.L.C. 12/2002 – current
71 Edgewood Way
Westville, CT 06515

Northeast Utilities /Dominion Resources Inc 12/1996 – 12/2002
(Under Contract via Cataract Inc through 9/97.)
2500 McClellan Ave.
Pennsauken, NJ 08109

New York Power Authority 11/1992 - 12/1996
123 Main Street
White Plains, New York 10671

Cygna Energy Services 11/1991 - 11/1992
5600 Glenridge Drive, Suite 380
Atlanta, Georgia 30075

ABB Impell Corporation 6/1978 - 11/1991
333 Research Court
Technology Park-Atlanta
Norcross, Georgia 30095

EDUCATION:

University of California, Berkeley
B.A. Physics, 1983
Senior level and graduate course work in Mechanical Engineering, and Electrical Engineering

Quinnipiac University School of Law
J.D expected June, 2009

PUBLICATIONS:

- EPRI Report Number 108736, "Guidelines for the Optimization of the Engineering Change Process," March 1994.
- NIRMA PP-03, "Position Paper for a Configuration Management Enhancement Program for a Nuclear Facility," April, 1992. Subcommittee Chair.
- EPRI Report Number 8480, " Seismic Piping Test and Analysis," 1980.

PROFESSIONAL AFFILIATIONS AND AWARDS

American Society of Mechanical Engineers, American Nuclear Society, Nuclear Information and Records Management Association, Who's Who For Rising Young Americans.

REFERENCES:

References available upon request.

EXHIBIT Q.1

EXHIBIT R

Leak found in pipe at Indian Point

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 vacuum test like the one that turned up the leak, as well as an ultrasonic test, will be performed to confirm the size and scope of the leak, Steets said. That will take a few more days. Repairs would follow, but would not require a reactor shutdown.

Plant officials say the leak has not contributed significantly to the groundwater contamination. The origin of the leak remains unclear.

"We'll know better about what might have caused it when we complete the testing that we're doing," Steets said. "You hate to speculate."

Nuclear Regulatory Commission spokesman Neil Sheehan said the leak was above where external moisture was found by workers during an excavation.

The leak point is under water only when the canal is flooded for refueling, which occurs every 18 to 24 months. More testing is needed before a connection can be drawn to the groundwater contamination, Sheehan said.

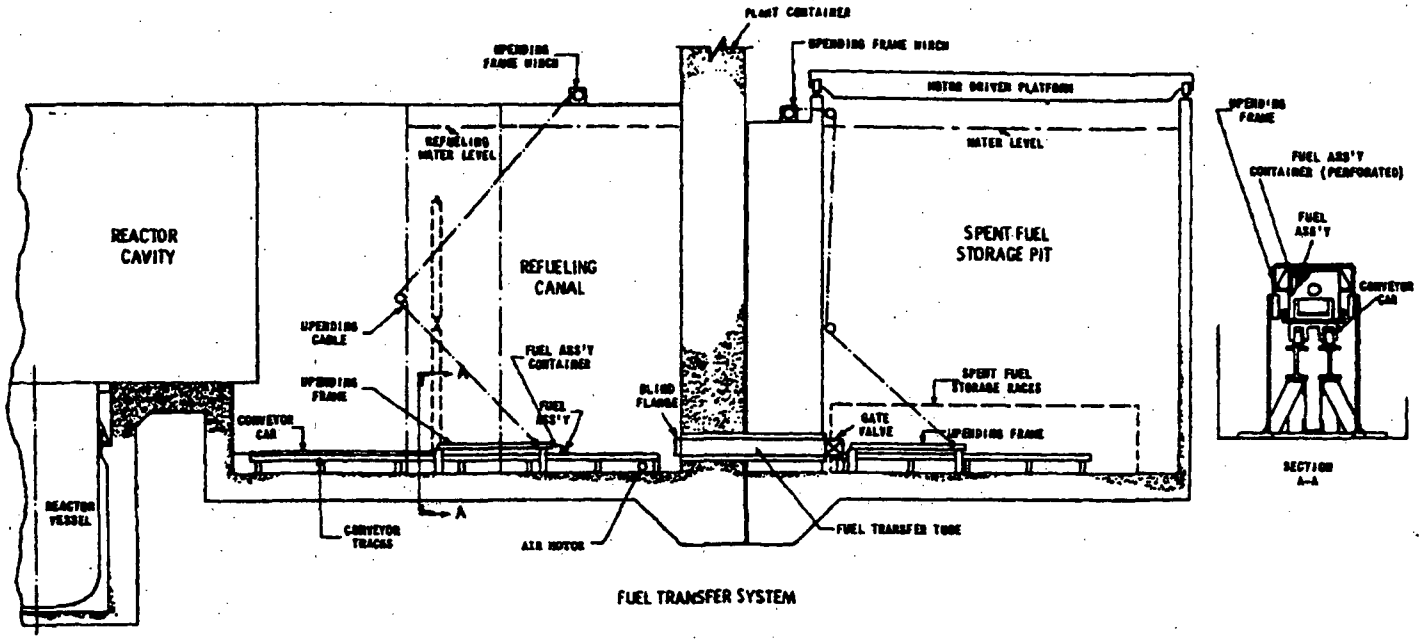
"Whether this is the cause, whether this is part of the cause, we don't know that yet, and there's still more work to be done," he said.

Buchanan Mayor Dan O'Neill learned of the leak yesterday and was assured there was no threat to the health of residents or workers at the plant.

"It does not sound like it's anything major at this time ... ." O'Neill said.

Phillip Musegaas, a staff attorney with the environmental group Riverkeeper, said the leak underscored why the NRC should require more thorough testing of systems holding radioactive water.

"This is a switch from Entergy's earlier position, because in their relicensing application they have stated that they didn't believe there was an ongoing leak at Indian Point 2 at all," Musegaas said. "The fact that they found this on further inspection suggests that they may find more leaks."



FUEL TRANSFER SYSTEM

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

EXHIBIT S

Entergy

Entergy Nuclear
P.O. Box 31995
Jackson, MS 39286-1995
Tel 601 368 5692

Michael R. Kansler
President, Chief Executive Officer
& Chief Nuclear Officer

July 30, 2007
ENOC-07-0026

U.S. Nuclear Regulatory Commission
Attention: James E. Dyer
Director, Office of Nuclear Reactor Regulation
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
Indian Point Nuclear Generating Unit No. 1
Docket No. 50-003
Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247
Indian Point Nuclear Generating Unit No. 3
Docket No. 50-286
James A. FitzPatrick Nuclear Power Plant
Docket Nos. 50-333 & 72-12
Vermont Yankee Nuclear Power Station
Docket Nos. 50-271
Palisades Nuclear Plant
Docket No. 50-255 & 72-7
Big Rock Point
Docket Nos. 50-155 & 72-43

Application for Order Approving Indirect Transfer of Control of Licenses

Pursuant to Section 184 of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.80, Entergy Nuclear Operations, Inc. (ENO), acting on behalf of itself and Entergy Nuclear Generation Company, Entergy Nuclear FitzPatrick, LLC, Entergy Nuclear Vermont Yankee, LLC, Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Palisades, LLC, (together, Applicants), hereby requests that the Nuclear Regulatory Commission (NRC) consent to the indirect transfer of control of the above-captioned licenses. The indirect transfer of control results from certain restructuring transactions that will involve the creation of new intermediary holding companies and/or changes in the intermediary holding companies for the ownership structure for the corporate entities that hold the NRC licenses for Pilgrim, Indian Point 1, 2, and 3, FitzPatrick, Vermont Yankee, Palisades and Big Rock Point (together, Facilities), including both the six corporate entities (named among the

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NMSS01
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Applicants above) licensed for their ownership of the Facilities and ENO, which is the entity licensed to operate or maintain the Facilities. The licensees remain the same, and the ultimate corporate parent, Entergy Corporation, remains the same. Simplified organization charts reflecting the current and post-reorganization ownership structures are provided as Figures 1 and 2.

Through the attached Application, ENO requests, on behalf of the Applicants, that the NRC consent to this proposed indirect transfer of control. The proposed indirect transfer of control will not result in any change in the role of ENO as the licensed operator of the facilities and will not result in any changes to its technical qualifications.

In summary, the proposed indirect transfer of control will be consistent with the requirements set forth in the Act, NRC regulations, and the relevant NRC licenses and orders. The proposed indirect transfer of control will not result in any physical changes to the Facilities or changes in the officers, personnel, or day-to-day operation of the Facilities. The proposed indirect transfer of control will not involve any changes to the current licensing basis of the Facilities. It will neither have any adverse impact on the public health and safety, nor be inimical to the common defense and security. This transfer does not involve any ownership, control or domination by any foreign entity. The Applicants therefore respectfully request that the NRC consent to the indirect transfer of control of the licenses for the Facilities in accordance with 10 CFR 50.80.

ENO requests that NRC review this Application on a schedule that will permit the issuance of NRC consent to the indirect transfer of control by December 31, 2007. Such consent should be made immediately effective upon issuance and should permit the indirect transfer of control at any time for one year following NRC's approval. ENO will inform NRC if there are any significant changes in the status of any other required approvals or any other developments that have an impact on the schedule.

The Application includes a proprietary, separately bound addendum that provides Attachments 2A and 3A, which contain confidential commercial or financial information. ENO requests that Attachments 2A and 3A be withheld from public disclosure pursuant to 10 CFR 2.390, as described in the Affidavit of Michael R. Kansler, which is provided in Attachment 4 to the Application. Non-proprietary versions of Attachments 2A and 3A suitable for public disclosure are provided as Attachments 2 and 3 to the Application.

Regulatory commitments made by Entergy are identified in the table provided in the Enclosure titled "Commitments ".

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If NRC requires additional information concerning this license transfer request, please contact John McCann, ENO's Director, Fleet Regulatory Affairs, at (914) 272-3370 or jmccan1@entergy.com. Service on ENO of comments, hearing requests or intervention petitions, or other pleadings, if applicable, should be made to counsel for ENO, Mr. John E. Matthews at Morgan, Lewis & Bockius, LLP, 1111 Pennsylvania Avenue, NW, Washington, DC 20004 (tel: 202-739-5524; fax: 202-739-3001; e-mail: jmatthews@morganlewis.com).

Michael R. Kansler
President & Chief Executive Officer

Enclosures: Regulatory Commitments
Oath & Affirmation
Application For Order Approving Indirect Transfer Of Control Of Licenses

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cc: w/o proprietary Addendum except *

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
2443 Warrenville Rd Suite 210
Lisle, IL 60532-4352

* U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
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Rockville, MD 20852

* Steven R. Hom
U. S. Nuclear Regulatory Commission
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Rockville, MD 20852
Mail Stop OWFN/12-D3

Mr. James J. Shea, Project Manager
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
Mail Stop 08 B1
Washington, DC 20555

Mr. John Boska, Project Manager
Plant Licensing Branch I-I
Division of Operating Reactor Licensing
Of.lce of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Mahesh L. Chawla, Project Manager
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
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Washington, DC 20555

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VT Department of Public Service
112 State Street - Drawer 20
Montpelier, Vermont 15620-2601

Mr. Peter R. Smith, President
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Department
of Public Service
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Albany, NY 12223

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
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New York, NY 10271

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Michigan Department of Attorney General
Special Litigation Division
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Sixth Floor, G. Mennen Williams Building
Lansing, MI 48913

Mr. Raymond L. Albanese
Four County Coordinator
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Unit 4 Westchester County
Hawthorne, NY 10532

USNRC Resident Inspector
Entergy Nuclear Vermont Yankee
P.O. Box 157
Vernon, Vermont 05354

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
James .A. FitzPatrick Nuclear Power Plant
PO Box 136
Lycoming, NY 13093

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Palisades Plant
27782 Blue Star Memorial Highway
Covert, MI 49043

Senior Resident Inspector's Office
Indian Point 2
U. S. Nuclear Regulatory Commission
P.O. Box 59
Buchanan, NY 10511

Senior Resident Inspector
Pilgrim Nuclear Power Station

Michigan Department of Environmental Quality
Waste and Hazardous Materials Division
Hazardous Waste and Radiological
Protection Section
Nuclear Facilities Unit
Constitution Hall, Lower-Level North
525 West Allegan Street
P.O. Box 30241
Lansing, MI 48909-7741

Mr. Robert Walker, Director
Massachusetts Department of Public Health
Schrafft Center Suite 1 M2A
Radiation Control Program
529 Main Street
Charlestown, MA 02129

Ms. Cristine McCombs, Director
Mass. Emergency Management Agency
400 Worcester Road
Framingham, MA 01702

Mr. Peter R. Smith, President
New York State Energy, Research,
& Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Supervisor
Covert Township
P. O. Box 35
Covert, MI 49043

Office of the Governor
P. O. Box 30013
Lansing, MI 48909

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Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
1. For entities listed on Attachment 1 that have not yet been formed, these entities will be formed in the states indicated, with the business address indicated, and with the Directors or Managers and Executive Personnel indicated.	x		No later than the date on which the indirect license transfers are implemented.
2. Entergy Nuclear Finance Holding, LLC, will execute a financial Support Agreement in favor of the Applicants substantially in the form provided in Attachment 5.	x		No later than the date on which the indirect license transfers are implemented.
3. Entergy Nuclear Finance Holding, LLC, will provide a letter of credit or other financial assurance instrument in compliance with 10 CFR 50.75(e)(1) to be held by Entergy Nuclear Palisades, LLC and to replace the $5 million Guaranty of decommissioning funding assurance for the Big Rock ISFSI.	x		No later than the date on which the indirect license transfers are implemented.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Entergy Nuclear Operations, Inc.	)	
	)	
Pilgrim Nuclear Power Station	)	Docket Nos. 50-293
Indian Point Nuclear Generating Unit No. 1	)	50-003
Indian Point Nuclear Generating Unit No. 2	)	50-247
Indian Point Nuclear Generating Unit No. 3	)	50-286
James A. FitzPatrick Nuclear Power Plant	)	50-333 &
FitzPatrick ISFSI	)	72-12
Vermont Yankee Nuclear Power Station	)	50-271
Palisades Nuclear Plant	)	50-255 &
Palisades ISFSI	)	72-7
Big Rock Point	)	50-155 &
Big Rock Point ISFSI	)	72-043

AFFIRMATION

I, Michael R. Kansler, being duly sworn, hereby depose and state: that I am President & Chief Executive Officer, of Entergy Nuclear Operations, Inc.; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached application for order approving indirect transfer of control of licenses; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

Michael R. Kansler
President & Chief Executive Officer

STATE OF MISSISSIPPI)
)
COUNTY OF HINDS)

Subscribed and sworn to me, a Notary Public, in and for the State of Mississippi, this 30th day of July, 2007.

Amy A. Blaylock
Notary Public in and for the State of Mississippi

Notary Public State of Mississippi At Large
My Commission Expires: June 17, 2009
Bonded Thru Helden, Brooks & Garland, Inc.

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Application for Order Approving Indirect Transfer of Control of Licenses

Entergy Nuclear Operations, Inc. (All Dockets)
Pilgrim Nuclear Power Station, Docket No. 50-293
Indian Point Nuclear Generating Unit No. 1, Docket No. 50-003
Indian Point Nuclear Generating Unit No. 2, Docket No. 50-247
Indian Point Nuclear Generating Unit No. 3, Docket No. 50-286
James A. FitzPatrick Nuclear Power Plant, Docket Nos. 50-333 & 72-12
Vermont Yankee Nuclear Power Station, Docket Nos. 50-271
Palisades Nuclear Plant, Docket No. 50-255 & 72-7
Big Rock Point, Docket Nos. 50-155 & 72-43

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Proprietary Addendum:

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I. INTRODUCTION

Pursuant to Section 184 of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.80, Entergy Nuclear Operations, Inc. (ENO), acting on behalf of itself and Entergy Nuclear Generation Company, Entergy Nuclear FitzPatrick, LLC, Entergy Nuclear Vermont Yankee, LLC, Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Palisades, LLC, (together, Applicants), hereby requests that the Nuclear Regulatory Commission (NRC) consent to the indirect transfer of control of the above-captioned licenses. The indirect transfer of control results from certain restructuring transactions that will involve the creation of new intermediary holding companies and/or changes in the intermediary holding companies for the ownership structure for the corporate entities that hold the NRC licenses for Pilgrim, Indian Point 1, 2, and 3, FitzPatrick, Vermont Yankee, Palisades and Big Rock Point (together, the Facilities), including both the six corporate entities (named among the Applicants above) licensed for their ownership of the Facilities and ENO, which is the entity licensed to operate and/or maintain the Facilities. The licensees remain the same, and the ultimate corporate parent, Entergy Corporation, remains the same. Simplified organization charts reflecting the current and post-reorganization ownership structures are provided as Figures 1 and 2.

II. STATEMENT OF PURPOSE OF THE TRANSFERS AND NATURE OF THE TRANSACTION MAKING THE TRANSFERS NECESSARY OR DESIRABLE

The restructuring transactions will centralize ownership and control of the owner Applicants under a new intermediate holding company structure in the Entergy Corporation system that will be wholly owned by Entergy Nuclear Finance Holding, LLC (HoldCo.). The transactions also will centralize ownership and control of ENO and Entergy's other nuclear service businesses under Entergy Nuclear, Inc. The restructuring will enhance the financial

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strength of the Applicants, simplify the Applicants' and Entergy Corporation's corporate structure to the benefit of customers, regulators, capital markets and shareholders, and facilitate the financing of Holdco and its direct and indirect subsidiaries as a discrete and integrated business. The restructuring is fully consistent with the continued safe operation of the Facilities. By reorganizing a currently diffuse organization, the wholesale nuclear business will be positioned for future growth.

For historic reasons the Applicants are currently part of a dispersed structure within the Entergy Corporation system. Financing is provided internally in a top down fashion, with debt attributable to the wholesale nuclear business residing primarily with Entergy Corporation. This structure has resulted in complex financing and operating relationships. The Applicants believe that by aggregating their ownership and financing activities under Holdco within a discrete business segment structure, and aggregating their nuclear services businesses under Entergy Nuclear, Inc., they will own and operate the company's nuclear plants with more clarity and enhance their ability to attract capital.

The restructuring will create an organizational structure that is consistent with Entergy Corporation's characterization and management of the wholesale, non-utility nuclear business as one of its primary business segments. Operating revenues and net income from its nuclear services business and its wholesale, non-utility nuclear generation business will be segregated for the benefit of this business segment. This will create discrete operating history and focused operating results.

The restructuring will isolate and simplify the structure of the businesses that comprise the wholesale nuclear business segment. This simplification will enhance the ability of analysts, regulators, capital markets and shareholders to understand and evaluate this business segment.

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The Applicants believe that the organization of a separate and integrated intermediate holding company system will clarify responsibilities within the Entergy Corporation system, facilitate capital formation, enhance the ability to retain and recruit qualified personnel and highlight growth opportunities for this important segment of Entergy Corporation's business.

III. GENERAL CORPORATE INFORMATION

The following are the names of the corporate entities licensed by the NRC:

Entergy Nuclear Operations, Inc.
Entergy Nuclear Generation Company
Entergy Nuclear FitzPatrick, LLC
Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Indian Point 2, LLC
Entergy Nuclear Indian Point 3, LLC
Entergy Nuclear Palisades, LLC

The following are the names of the parent corporate entities that will directly or indirectly own the NRC licensed corporate entities.

Entergy Corporation
Entergy Nuclear, Inc.
(by merger, successor to Entergy Nuclear Holding Company #2)
Entergy Global Trading Holdings, LTD
Entergy International Holdings, LTD
Entergy Global Investments, Inc.
(formerly, Entergy Global, LLC)
Entergy Power Gas Holdings Corp.
Entergy Power Gas Operations Corp.
Entergy Nuclear Holding Company #1
Entergy Global Holdings, Inc.
Entergy Nuclear Finance Holding, LLC
(formerly, Entergy Nuclear Finance Holding, Inc.)
Entergy Nuclear Holding, LLC
(formerly, Entergy Nuclear Holding Company)
Entergy NHC, LLC
Entergy Nuclear Midwest Investment Company, LLC
Entergy Nuclear Northeast Investment Company, LLC
(formerly, Entergy Nuclear New York Investment Company 1, and
by merger, successor to Entergy Nuclear Holding Company #3 LLC)
Entergy Nuclear Investment Company, LLC
Entergy Nuclear Vermont Investment Company, LLC

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The parent company relationships of the licensed corporate entities both before and after the indirect transfer of control are reflected in Figures 1 and 2. The information regarding each corporate entity required by 10 CFR 50.33(d)(3) is provided in Attachment 1.

All of the current and proposed directors and executive personnel of the corporate entities are citizens of the United States.

IV. FOREIGN OWNERSHIP OR CONTROL

Entergy Corporation is a publicly traded company, and its securities are traded on the New York Stock Exchange and are widely held. Section 13(d) of the Securities Exchange Act of 1934, as amended, 15 U.S.C. 78m(d), requires that a person or entity that owns or controls more than 5% of the securities of a company must file notice with the Securities and Exchange Commission (SEC). Based upon filings with the SEC, ENO is aware of one alien, foreign corporation, or foreign government that holds or may hold beneficial ownership of more than 5% of the securities of Entergy Corporation. AXA Assurance I.A.R.D. Mutuelle, a French entity, and its affiliates (together, AXA) have filed a statement indicating that as of December 31, 2006, AXA had beneficial ownership of 5% of the shares of Entergy Corporation. AXA does not have any representation on Entergy Corporation's Board of Directors, and its SEC filing specifically certifies that AXA did not acquire these shares for the purpose of or with the effect of changing or influencing the control of Entergy Corporation. See 17 CFR 240.13d-1(c)(1) (requirements for Schedule 13G filing).

The current and proposed directors and executive officers of Entergy Corporation and the Entergy subsidiaries that directly or indirectly own the Applicants are United States citizens. There is no reason to believe that the Applicants are owned, controlled, or dominated by any alien, foreign corporation, or foreign government. Thus, the indirect transfer of control of the

licensed entities and their corporate parents will not result in any foreign ownership, domination, or control of these entities within the meaning of the Atomic Energy Act of 1954, as amended.

V. TECHNICAL QUALIFICATIONS

The technical qualifications of ENO are not affected by the proposed indirect transfer of control. There will be no physical changes to the Facilities and no changes in the officers, personnel, or day-to-day operations of the Facilities in connection with the indirect transfer of control. It is anticipated that ENO will at all times remain the licensed operator of the Facilities, or in the case of permanently shutdown reactors the entity licensed to maintain the Facilities.

VI. FINANCIAL QUALIFICATIONS

The Applicants are all indirect, wholly-owned subsidiaries of Entergy Corporation ("Entergy"). Headquartered in New Orleans, Louisiana, Entergy is an integrated energy company engaged primarily in electric power production and retail electric distribution operations. Entergy owns and operates power plants with approximately 30,000 MW of electric generating capacity, and Entergy is the second-largest nuclear power generator in the United States. Entergy delivers electricity to 2.6 million utility customers in Arkansas, Louisiana, Mississippi, and Texas. Entergy generated annual revenues of $10.9 billion in 2006 and had approximately 13,800 employees as of December 31, 2006. Through its subsidiaries (both regulated and non-regulated), Entergy Corporation owns and operates eleven nuclear power plants at nine sites. These include the Facilities that are the subject of this application, as well as five other nuclear power plants owned by affiliates of the Applicants: Arkansas Nuclear One Units 1 and 2, Grand Gulf Nuclear Station, River Bend Station, and Waterford 3 Steam Electric Station.

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A. Projected Operating Revenues and Operating Costs

Financial information regarding Entergy Corporation and its subsidiaries is provided in its 2006 Annual Report (SEC Form 10-K) dated March 1, 2007, which is available along with Entergy's prior annual reports on the internet at:

http://www.shareholder.com/entergy/edgar.cfm?DocType=Annual,Quarterly&Year=

In addition, Applicants have prepared balance sheets and projected income statements for the licensed owners of the Facilities, as well as a projected consolidated balance sheet and projected income statement for Entergy Nuclear Finance Holding, LLC (HoldCo), which is an intermediary holding company that will indirectly own all of the corporate entities licensed to own the Facilities, as well as other assets and businesses related to non-utility nuclear generation business of Entergy Corporation.

ENO, the corporate entity licensed to operate the operating Facilities and to maintain the non-operating Facilities, will be a wholly-owned subsidiary of Entergy Nuclear, Inc., which itself will be a direct wholly-owned subsidiary of Entergy Corporation. Entergy Nuclear, Inc, will own the nuclear services businesses of Entergy Corporation. ENO will receive the revenue necessary to operate and maintain the Facilities, including decommissioning funds to pay for such expenses, from the corporate entities licensed to own the Facilities pursuant to operating agreements or other intra-corporate arrangements that have been previously described to NRC. If any changes are made to replace the existing arrangements, any new agreements are expected to be consistent with the current arrangements. Any new agreements will be made available for inspection by NRC. As such, ENO relies upon the financial qualifications of the licensed owners of the Facilities, because these corporate entities will be financially responsible for the operation and decommissioning of the units.

In accordance with 10 CFR 50.33(f) and the Standard Review Plan on Power Reactor Licensee Financial Qualifications and Decommissioning Funding Assurance (NUREG-1577, Rev. 1) ("Standard Review Plan"), projected balance sheets for each of the licensed owners of the Facilities are provided in a separately bound proprietary addendum as Attachment 2A. In addition, a projected opening balance sheet for the consolidated businesses of Entergy Nuclear Finance Holding, LLC is also provided in Attachment 2A. ENO requests that Attachment 2A be withheld from public disclosure, as described in the Affidavit provided in Attachment 4. Redacted versions of these balance sheets, suitable for public disclosure, are provided as Attachment 2.

In addition, *pro forma* Projected Income Statements for the six year period from January 1, 2007 through December 31, 2012 for each of the licensed owners of the Facilities and Entergy Nuclear Finance Holding, LLC are provided in a separately bound proprietary addendum as Attachment 3A. In addition, a sensitivity analysis of these projections (reflecting a 10% reduction in projected revenue) is provided in Attachment 3A. ENO requests that Attachment 3A be withheld from public disclosure, as described in the Affidavits provided in Attachment 4. Redacted versions of these balance sheets, suitable for public disclosure, are provided as Attachment 3.

The Projected Income Statements for the licensed owners show that anticipated revenues from sales of capacity and energy from the Facilities provide reasonable assurance of an adequate source of funds to meet the ongoing operating and maintenance expenses for the Facilities. In addition, Entergy Nuclear Finance Holding, LLC will execute a financial Support Agreement with the Applicants, including each of the corporate entities licensed to own the Facilities, in the total amount of $700 million, to pay for the operating and maintenance (O&M)

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costs for all six operating Facilities, if called upon to do so. This provides assurance that adequate funds will be available to fund ongoing O&M expenses with respect to all of the operating Facilities. A form of this agreement is provided as Attachment 5.

The financial projections for Entergy Nuclear Finance Holding, LLC establish that it will have adequate resources from its consolidated businesses to provide funding if necessary under the Support Agreement. In addition, this parent company is expected to have access to a line of credit of at least $1 billion or more, which provides additional assurance of its ability on an ongoing basis to provide funds for the licensed entities.

Pursuant to the Support Agreement, the licensed owners will have access to funds sufficient to pay the fixed O&M costs in the event of any unanticipated plant shutdown in accordance with the guidance provided in the Standard Review Plan. Pursuant to this agreement, Entergy Nuclear Finance Holding, LLC will make up to an aggregate amount of $700 million in funding available to any and all of the Applicants to meet their obligations to NRC relating to the Facilities. This arrangement replaces the prior financial support arrangements under which funds were available to each licensed owner individually in limited amounts, and Applicants seek NRC's prior written approval of the revocation of the prior arrangements through NRC's approval of the new Support Agreement, which rescinds the prior arrangements under the terms of Section 7 of the Support Agreement.

Under the new Support Agreement, each of the licensed entities will have access to up to a total of $700 million, to the extent not previously utilized, for any single plant outage or for a multiple plant outage should the circumstances necessitate access to such funds. As such, the proposed Support Agreement would provide funding for any individual site that significantly exceeds the six-month period suggested by the NRC's Standard Review Plan guidance, which

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requests demonstration of a source of funds to pay fixed O&M expenses in the event of an extended plant outage. The availability of the entire aggregate amount of funding under the Support Agreement for each plant is superior to the current disparate support arrangements. Moreover, the total amount available would fund nearly six-months worth of fixed O&M expenses for all six operating Facilities. Finally, Applicants note that they do not expect to need to request funding under this formal agreement, as they expect that during their day-to-day operations and otherwise as the need for funding arises, they will have access to funds from capital contributions, loans, credit lines, or other sources that provide adequate funding to support safe operation of all of the Facilities.

B. Decommissioning Funding

The financial qualifications of the Applicants to continue to own the Facilities are further demonstrated by the decommissioning funding assurance provided in accordance with 10 CFR 50.75(e)(1). Details regarding the status of the decommissioning funding assurance maintained by the Applicants for the Facilities are provided in the March 29, 2007 decommissioning funding status report (ENOC-07-00007) submitted by ENO in accordance with 10 CFR 50.75(f), except for Palisades and Big Rock Point which were not included in this report. This report demonstrates that there is reasonable assurance of adequate decommissioning funding that is provided by pre-paid amounts maintained as assets in external sinking funds segregated from licensee assets and outside licensee administrative control in accordance with the requirements of 10 CFR 50.75(e)(1)(i).

With respect to Palisades, the trust fund balance for Palisades as of April 30, 2007 was approximately $252.9 million, and with credit for earnings taken into account as permitted by NRC rules, less than $205 million in pre-paid assets maintained in a trust would be sufficient to fully fund the NRC's current "formula amount" estimate for Palisades decommissioning costs at

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$345.9 million, calculated pursuant to 10 CFR 50.75(c). Thus, the existing trust fund balances maintained by Entergy Palisades LLC as assets in an external sinking fund segregated from licensee assets and outside licensee administrative control provide decommissioning funding assurance in accordance with the requirements of 10 CFR 50.75(e)(1)(i). There is, therefore, reasonable assurance that the amount of decommissioning funds available will be sufficient to pay decommissioning costs for Palisades at the time permanent termination of operations is expected.

With respect to Big Rock Point, the NRC acknowledged in its recent approval of the transfer of this facility to Entergy Palisades LLC that NRC has approved the release of most of the Big Rock Point site, and the remaining decommissioning obligation is approximately $2.8 million estimated for the decommissioning of the Independent Spent Fuel Storage Facility (ISFSI). Entergy Corporation committed to provide a Parent Guaranty for $5 million. Prior to the indirect transfer of the Big Rock Point license, this Parent Guaranty will be terminated and replaced by an alternative financial assurance mechanism acceptable under the terms of 10 CFR 50.75(e)(1), such as a letter of credit from a financial institution or a pre-paid decommissioning trust in an amount not less than $2.8 million. None of the other existing arrangements for Big Rock Point as approved in the prior license transfer will be affected. This provides reasonable assurance of the availability of funds for decommissioning the Big Rock Point ISFSI pursuant to 10 CFR 50.75 and 72.30.

Other than the changes to the Parent Guaranty for Big Rock Point described above, the Applicants do not anticipate any changes in the existing decommissioning funding assurance provided in connection with the proposed indirect transfers of control. Applicants also do not anticipate any changes or amendments to any nuclear decommissioning trust fund agreements,

and if any amendments are to be made in the future, the existing trust agreements require prior written notice be provided to the NRC. Moreover, any existing NRC license conditions governing these trust agreements will remain in effect and unchanged.

VII. ANTITRUST INFORMATION

This Application post-dates the issuance of the operating licenses of the facilities, and therefore no antitrust review is required or authorized. Based upon the Commission's decision in *Kansas Gas and Electric Co., et al.* (Wolf Creek Generating Station, Unit 1), CLI-99-19, 49 NRC 441 (1999), the Atomic Energy Act of 1954, as amended, does not require or authorize antitrust reviews of post-operating license transfer applications.

VIII. RESTRICTED DATA AND CLASSIFIED NATIONAL SECURITY INFORMATION

The proposed transfers do not involve any Restricted Data or other Classified National Security Information or result in any change in access to such Restricted Data or Classified National Security Information. ENO's existing restrictions on access to Restricted Data and Classified National Security Information are unaffected by the proposed transfers. In compliance with Section 145(a) of the Act, the applicants agree that restricted or classified defense information will not be provided to any individual until the Office of Personnel Management investigates and reports to the NRC on the character, associations, and loyalty of such individual, and the NRC determines that permitting such person to have access to Restricted Data will not endanger the common defense and security of the United States.

IX. ENVIRONMENTAL CONSIDERATIONS

The requested consent to indirect transfer of control of the facilities' licenses is exempt from environmental review because it falls within the categorical exclusion contained in 10 CFR 51.22(c)(21), for which neither an Environmental Assessment nor an Environmental Impact

Statement is required. Moreover, the proposed indirect transfer does not involve any amendment to the facility operating licenses or other change, and it will not directly affect the actual operation of the Facilities in any substantive way. The proposed transfer does not involve an increase in the amounts, or a change in the types, of any radiological effluents that may be allowed to be released off-site, and involves no increase in the amounts or change in the types of non-radiological effluents that may be released off-site. Further, there is no increase in the individual or cumulative operational radiation exposure, and the proposed transfer has no environmental impact.

X. PRICE-ANDERSON INDEMNITY AND NUCLEAR INSURANCE

The proposed indirect transfer of control does not affect the existing Price-Anderson indemnity agreements for the Facilities, and does not affect the required nuclear property damage insurance pursuant to 10 CFR 50.54(w) and nuclear energy liability insurance pursuant to Section 170 of the Act and 10 CFR Part 140.

XI. EFFECTIVE DATE AND OTHER REQUIRED REGULATORY APPROVALS

Accordingly, ENO requests that NRC review this Application on a schedule that will permit the issuance of NRC consent to the indirect transfer of control by December 31, 2007. Such consent should be made immediately effective upon issuance and should permit the indirect transfer of control at any time within a year after issuance. ENO will inform the NRC if there are any significant changes in the status of any other required approvals or any other developments that have an impact on the schedule.

XII. CONCLUSION

Based upon the foregoing information, ENO respectfully requests, on behalf of the Applicants, that the NRC issue an Order consenting to the indirect transfer of control.

FIGURE 1

SIMPLIFIED ORGANIZATION CHART – CURRENT

Figure 1: SIMPLIFIED ORGANIZATION CHART – CURRENT

NOTE: NRC Licenses held by E N entities are referenced in parentheses and italics, *e.g.*, **(IP1 & IP2)**.

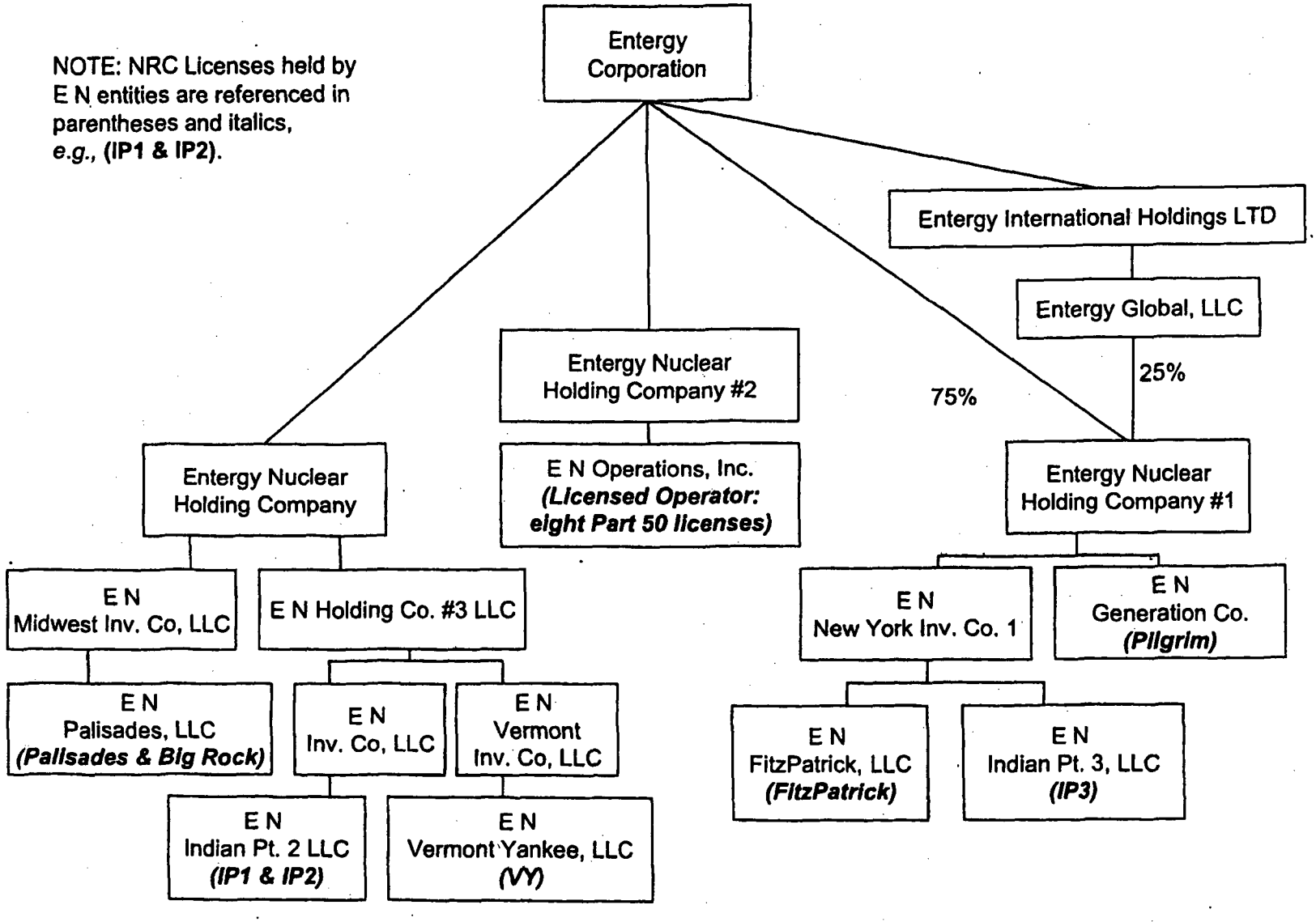

FIGURE 2

SIMPLIFIED ORGANIZATION CHART – POST REORGANIZATION

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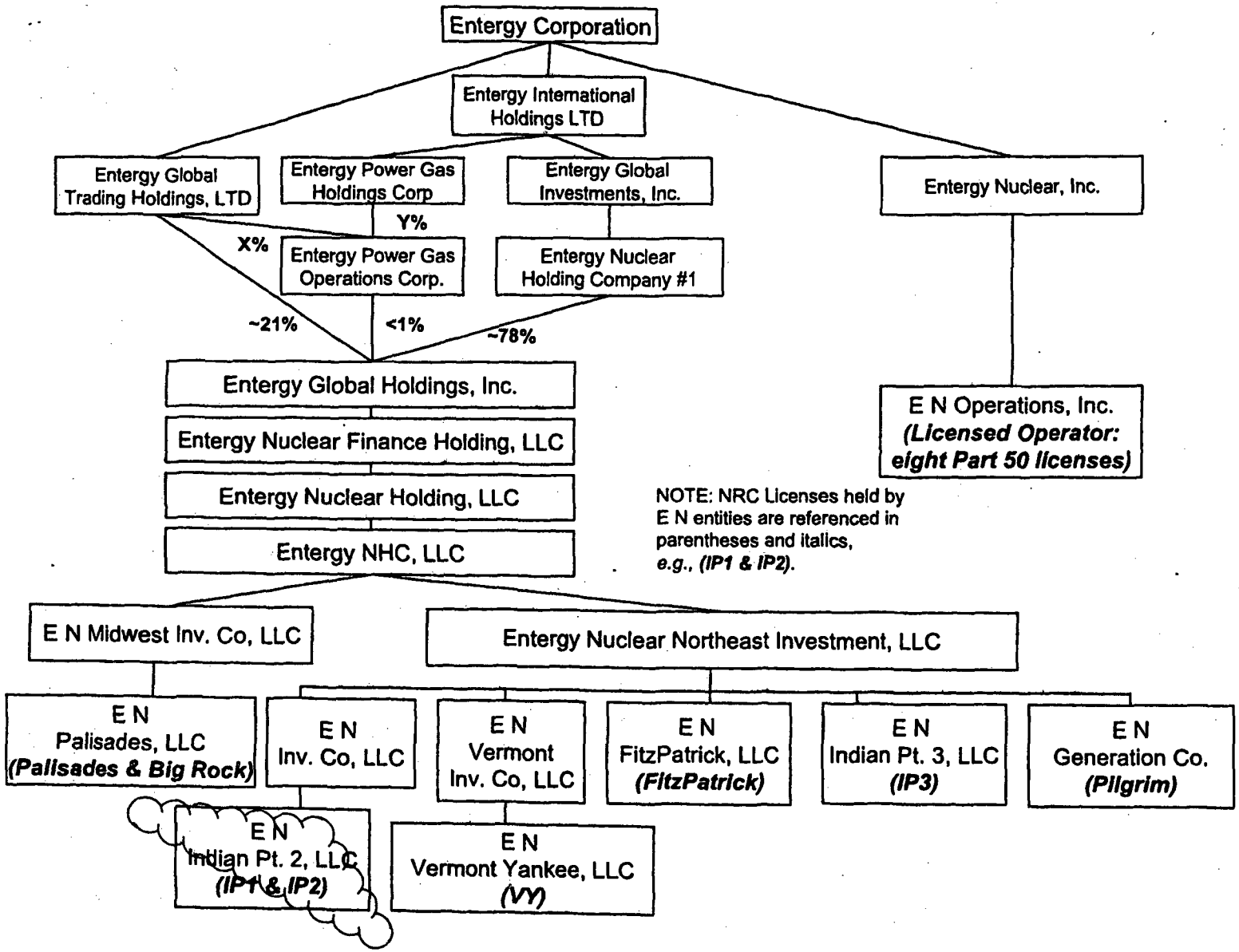
Figure 2: SIMPLIFIED ORGANIZATION CHART – POST REORGANIZATION



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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Corporation
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	639 Loyola Avenue New Orleans, LA 70113
DIRECTORS:	J. Wayne Leonard (Chairman) Maureen S. Bateman W. Frank Blount Simon D. deBree Gary W. Edwards Alexis M. Herman Donald C. Hintz Stuart L. Levinick James R. Nichols William A. Percy, II W. J. "Billy" Tauzin Steven V. Wilkinson
EXECUTIVE PERSONNEL	J. Wayne Leonard – Chief Executive Officer Richard J. Smith – President & Chief Operating Officer Gary J. Taylor – Group President, Utility Operations Leo P. Denault – Executive VP & CFO Curtis L. Hebert, Jr. – Executive VP, External Affairs Michael R. Kansler – Executive VP & Chief Nuclear Officer Mark T. Savoff – Executive VP, Operations Robert D. Sloan – Executive VP / General Counsel & Secretary Theodore H. Bunting, Jr – Senior VP & Chief Accounting Officer Joseph T. Henderson – Senior VP & General Tax Counsel Terry R. Seamons – Senior VP, Human Resources & Administration Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear, Inc.
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS:	Michael R. Kansler – Chairman Leo P. Denault C. Randy Hutchinson
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer C. Randy Hutchinson – Senior VP, Development Robert D. Sloan – Executive VP & Secretary Wanda Curry – VP Chief Financial Officer, Nuclear Operations Danny R. Keuter – VP Business Development Steven C. McNeal – VP & Treasurer Dana Atchison – Assistant Secretary Amy A. Blaylock – Assistant Secretary Terence A. Burke – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Global Trading Holdings, LTD
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	10055 Grogans Mill Road, Parkwood II Building The Woodlands, TX 77380
DIRECTORS:	Barrett E. Green John Wengler James E. Striedel
EXECUTIVE PERSONNEL	Barrett E. Green – President John Wengler – VP & Treasurer James E. Striedel – Vice President Thomas Wagner – Secretary Joseph T. Henderson – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy International Holdings LTD
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, TX 77046
DIRECTORS:	Steven C. McNeal Eddie Peebles Andrew Rosenlieb
EXECUTIVE PERSONNEL	Eddie Peebles – President Steven C. McNeal – Vice President & Treasurer Andrew Rosenlieb – Vice President & Secretary Thomas G. Wagner – Assistant Secretary Joseph T. Henderson – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Global Investments, Inc. (Proposed Conversion)
STATE OF INCORPORATION:	Arkansas
BUSINESS ADDRESS:	425 West Capitol Avenue Little Rock, AR 72201
DIRECTORS:	Douglas Castleberry Steven C. McNeal O. H. Storey, III
EXECUTIVE PERSONNEL	Douglas Castleberry – President Robert D. Sloan – Executive VP, General Counsel, & Secretary Steven C. McNeal – Vice President & Treasurer O. H. Storey, III – Vice President Sue Chambers – Assistant Secretary Janan E. Honeysuckle – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Power Gas Holdings Corporation
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, Texas 77046
DIRECTORS:	Steven C. McNeal
EXECUTIVE PERSONNEL	James E. Striedel – President Joseph T. Henderson – Tax Officer Steven C. McNeal – VP & Treasurer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Power Gas Operations Corporation
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	Entity Services (Nevada), L.L.C. 2215-B Renaissance Dr., Suite 5 Las Vegas Nevada 89119
DIRECTORS:	Richard F. Boland Douglas Castleberry Steven C. McNeal Tom D. Reagan
EXECUTIVE PERSONNEL	Tom D. Reagan – President Richard F. Boland – VP, Secretary, & Assistant Treasurer Steven C. McNeal – VP & Treasurer Thomas G. Wagner – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Holding Company #1
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, Texas 77046
DIRECTORS:	Michael R. Kansler (Chairman) Wanda Curry
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer Joseph T. Henderson – Senior VP & General Tax Counsel Wanda Curry – VP Thomas G. Wagner – Secretary Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer Steven C. McNeal – VP & Treasurer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Global Holdings, Inc. (Proposed Entity/Not Yet Created)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	10055 Grogans Mill Road, Parkwood II Building The Woodlands, TX 77380
DIRECTORS:	James E. Striedel* Andrew Rosenlieb*
EXECUTIVE PERSONNEL	James E. Striedel* – President Andrew Rosenlieb* – Vice President John Wengler* – VP & Treasurer Reginald G. Rice* – Secretary Joseph C. Henderson* – Tax Officer

*Subject to additional internal review by Affiliate Rules Compliance

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Finance Holding, LLC (Proposed Conversion)
STATE OF INCORPORATION:	Arkansas
BUSINESS ADDRESS:	425 West Capitol Little Rock, AR 72201
MANAGERS:	Douglas Castleberry – Management Committee Member Michael R. Kansler – Management Committee Member O. H. Storey – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer Douglas Castleberry – Vice President Steven C. McNeal – VP & Treasurer O. H. Storey – VP & Secretary Sue Chambers – Assistant Secretary Janan E. Honeysuckle – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Holding, LLC (Proposed Conversion)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	20 Greenway Plaza, Suite 500 Houston, Texas 77046
MANAGERS:	Wanda Curry – Management Committee Member Eddie Peebles – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President & Chief Executive Officer Robert D. Sloan – Executive VP & Secretary Joseph T. Henderson – Senior VP & General Tax Counsel Wanda Curry – Vice President Steven C. McNeal – VP & Treasurer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy NHC, LLC (Proposed Entity/Not Yet Created)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	10055 Grogans Mill Road, Parkwood II Building The Woodlands, TX 77380
MANAGERS:	James E. Striedel* Andrew Rosenlieb*
EXECUTIVE PERSONNEL	James E. Striedel* – President Andrew Rosenlieb* – Vice President John Wengler* – VP & Treasurer Reginald G. Rice* – Secretary Joseph C. Henderson* – Tax Officer

* Subject to additional internal review by Affiliate Rules Compliance

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Midwest Investment Company, LLC
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
MANAGERS:	C. Randy Hutchinson – Management Committee Member
EXECUTIVE PERSONNEL	Joseph T. Henderson – Senior VP & General Tax Counsel Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary David Gibbs – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Northeast Investment Company, LLC (Proposed Conversion)
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS OR MANAGERS:	Michael R. Kansler – Management Committee Member C. Randy Hutchinson – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President, Executive VP & Chief Executive Officer Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Investment Company, LLC
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
MANAGERS:	C. Randy Hutchinson – Management Committee Member Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Terence A. Burke – VP & Secretary Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Vermont Investment Company, LLC
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
MANAGERS:	C. Randy Hutchinson – Management Committee Member Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Terence A. Burke – VP & Secretary Paul A. Castanon – Assistant Secretary Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Operations, Inc. [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS:	C. Randy Hutchinson Michael R. Kansler
EXECUTIVE PERSONNEL	Michael R. Kansler –Chief Executive Officer John McGaha – President, Planning, Development & Oversight John T. Herron – Senior VP, Entergy Nuclear Operations C. Randy Hutchinson – Senior VP, Business Development Robert D. Sloan – Executive VP, General Counsel & Secretary Michael A. Balduzzi, Jr. – Senior VP, Chief Operating Officer, ENO Kevin Bronson – VP Operations, Pilgrim Wanda Curry – VP, Chief Financial Officer, Nuclear Fred R. Dacimo – VP Operations, Indian Point Energy Center Peter T. Dietrich – VP Operations, JAF Danny R. Keuter – VP, Development, Planning & Innovation Oscar Limpias – VP, Engineering Steven C. McNeal – VP & Treasurer Stewart B. Minahan – VP Operations, Cooper Christopher J. Schwarz – VP Operations, Palisades Theodore A. Sullivan – VP Operations, Vermont Yankee Amy A. Blaylock – Assistant Secretary Terence A. Burke – Assistant Secretary Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer Paul Hinnenkamp – VP, Business Development Cliff Eubanks – VP, Project Management Joseph DeRoy – VP, Operations Support Bruce Williams – VP, Oversight

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Generation Company [NRC Licensed Entity]
STATE OF INCORPORATION:	Massachusetts
BUSINESS ADDRESS:	1340 Echelon Parkway Jackson, Mississippi 39213
DIRECTORS:	Michael R. Kansler – Chairman C. Randy Hutchinson
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President Robert D. Sloan – Executive VP & Secretary John T. Herron – Senior VP & Chief Operating Officer Michael A. Balduzzi, Jr. – VP, Operations, Pilgrim NPS Wanda Curry – VP, Chief Financial Officer, Nuclear Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary James W. Snider – Assistant Secretary Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear FitzPatrick, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	268 Lake Road East Lycoming, New York 13093
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President John T. Herron – Senior VP & Chief Operating Officer Robert D. Sloan – Executive VP, General Counsel & Secretary Wanda Curry – VP, Chief Financial Officer, Nuclear Peter T. Dietrich – VP, Operations Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Vermont Yankee, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	320 Governor Hunt Road Vernon, Vermont 05302
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President Robert D. Sloan – Executive VP, General Counsel & Secretary John T. Herron – Senior VP & Chief Operating Officer Wanda Curry – Vice President, Chief Financial Officer, Nuclear Operations Steven C. McNeal – VP & Treasurer Theodore A. Sullivan – VP, Operations Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Indian Point 2, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	Bleakley Avenue and Broadway Buchanan, New York 10511
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President Robert D. Sloan – Executive VP & Secretary John T. Herron – Senior VP & Chief Operating Officer Wanda Curry – VP, Chief Financial Officer, Nuclear Operations Fred R. Dacimo – Vice President, Operations Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Indian Point 3, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	Bleakley Avenue and Broadway Buchanan, New York 10511
MANAGERS:	Michael R. Kansler – Management Committee Member
EXECUTIVE PERSONNEL	Michael R. Kansler – Chief Executive Officer & President John T. Herron – Senior VP & Chief Operating Officer Robert D. Sloan – Executive VP, General Counsel & Secretary Wanda Curry – VP, Chief Financial Officer, Nuclear Operations Fred R. Dacimo – Vice President, Operations Steven C. McNeal – VP & Treasurer Paul A. Castanon – Assistant Secretary Mary Ann Valladares – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 1

General Corporate Information Regarding the NRC Licensed Entities and Their Corporate Parents

NAME:	Entergy Nuclear Palisades, LLC [NRC Licensed Entity]
STATE OF INCORPORATION:	Delaware
BUSINESS ADDRESS:	27780 Blue Star Memorial Highway Covert, Michigan 49043
MEMBER (MEMBER MANAGED LLC):	Entergy Nuclear Midwest Investment Company, LLC – Member
EXECUTIVE PERSONNEL	Michael R. Kansler – President Terence A. Burke – VP & Secretary Steven C. McNeal – VP & Treasurer Christopher J. Schwartz – VP, Operations Dana Atchison – Assistant Secretary Amy A. Blaylock – Assistant Secretary Paul A. Castanon – Assistant Secretary David Gibbs – Assistant Treasurer Patricia A. Galbraith – Tax Officer Rory L. Roberts – Tax Officer

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ATTACHMENT 2

Projected Balance Sheets: 2007-2012

(Non-Proprietary Version).

Entergy Nuclear Finance Holding, LLC (Consolidated) -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability and Other						
Notes Payable (1)						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity (1)						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 2, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands

Forecast as of April 2007	Projected Balance as of December 31					
	2007	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 3, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant (1)						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear FitzPatrick, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant (1)						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Generation Company -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Palisades, LLC -- Projected Balance Sheets (2007-2012)

Dollars in Thousands	Projected Balance as of December 31					
Forecast as of April 2007	**2007**	**2008**	**2009**	**2010**	**2011**	**2012**
ASSETS:						
Cash						
Accounts Receivable						
Fuel						
Inventory						
Notes Receivable						
Net Plant						
Decommissioning Trust Funds						
Prepayments & Other						
Total Assets						
LIABILITIES:						
Accounts Payable						
Accum. Def. Income Taxes						
Accrued Pension Liability and Other						
Notes Payable						
Decommissioning Liability						
Other Liabilities						
Total Liabilities						
EQUITY:						
Member's Interest						
Retained Earnings						
Total Equity						
Total Liabilities & Equity						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

ATTACHMENT 3

Projected Income Statements: 2007-2012

(Non-Proprietary Version)

Entergy Nuclear Finance Holding, LLC (Consolidated) -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Entergy Nuclear MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 2, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Indian Point 2 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
IP-1						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 3, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
IP3 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Vermont Yankee MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income (1)						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Cash Flow Statements (2007-2012)
Dollars in Thousands
Forecast as of April 2007

	2007	2008	2009	2010	2011	2012
OPERATING ACTIVITIES						
Net Income						
Non-Cash Items Included in Net Income:						
Depreciation, Amortization, Decommissioning and Deferred Income Taxes						
Other						
NET CASH FLOW PROVIDED BY (USED IN) OPERATING ACTIVITIES						
INVESTING ACTIVITIES						
Construction Expenditures						
Nuclear Fuel Purchase						
Decommissioning Trust Contributions and Realized Changes in Trust Assets						
NET CASH FLOW PROVIDED BY (USED IN) INVESTING ACTIVITIES						
FINANCING ACTIVITIES						
Proceeds from Issuance of:						
Long-Term Debt						
Retirement of:						
Long-Term Debt						
Notes from Parents / Associated Companies						
Other						
NET CASH FLOW PROVIDED BY (USED IN) FINANCING ACTIVITIES						
Net Increase (Decrease) in Cash and Cash Equivalents						
Cash and Cash Equivalents at Beginning of Period						
CASH AND CASH EQUIVALENTS AT END OF PERIOD						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear FitzPatrick, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Fitzpatrick MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Generation Company -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Pilgrim MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Palisades, LLC -- Projected Income Statements (2007-2012)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Palisades MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Big Rock ISFI						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Finance Holding, LLC (Consolidated) -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Entergy Nuclear MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 2, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Indian Point 2 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
IP-1						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						

Note: Assumes 01/01/08 Close

Total Operating Expenses

Add:
Ongoing Capital Expenditures

Less:
Plant Depreciation
Variable Outside Goods & Services
(25% of 25% of O&M)
Fuel
Outage

Annual Fixed Operating Expenses
6 Months' Operating Expenses

FORECASE STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Indian Point 3, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
IP3 MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Vermont Yankee, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)

Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Vermont Yankee MDC	605	605	605	605	605	605
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear FitzPatrick, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Fitzpatrick MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Generation Company – Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Pilgrim MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Entergy Nuclear Palisades, LLC -- Projected Income Statements (2007-2012)
Sensitivity (10% Reduction in Revenue)
Dollars in Thousands

Forecast as of April 2007	2007	2008	2009	2010	2011	2012
Palisades MDC						
Projected Capacity Factor						
Average Contract Price $/MWh						
Average Market Price $/MWh						
Power Sales - Contract						
Power Sales - Market						
Total Revenue						
Operation & Maintenance						
O&M						
Outage						
Insurance						
Other						
Big Rock ISFI						
Fuel						
DOE Charges						
Amortization						
Plant Depreciation						
Other						
Interest Income						
Interest Expense						
Decommissioning						
Administrative & Other						
Total Operating Expenses						
Operating Profit						
Income Taxes						
Net Income						
Note: Assumes 01/01/08 Close						
Total Operating Expenses						
Add:						
Ongoing Capital Expenditures						
Less:						
Plant Depreciation						
Variable Outside Goods & Services (25% of 25% of O&M)						
Fuel						
Outage						
Annual Fixed Operating Expenses						
6 Months' Operating Expenses						

FORECAST STATEMENTS / ACTUAL RESULTS MAY VARY

Attachment 4

10 CFR 2.390 AFFIDAVIT OF MICHAEL R. KANSLER

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Entergy Nuclear Operations, Inc.	)	
	)	
Pilgrim Nuclear Power Station	)	Docket Nos. 50-293
Indian Point Nuclear Generating Unit No. 1	)	50-003
Indian Point Nuclear Generating Unit No. 2	)	50-247
Indian Point Nuclear Generating Unit No. 3	)	50-286
James A. FitzPatrick Nuclear Power Plant	)	50-333
Vermont Yankee Nuclear Power Station	)	50-271
Palisades Nuclear Plant	)	50-255

AFFIDAVIT

I, Michael R. Kansler, President & Chief Executive Officer of Entergy Nuclear Operations, Inc. (ENO), hereby affirm and state:

1. I am authorized to execute this affidavit on behalf of ENO.

2. ENO is providing information in support of an Application for Order Approving Indirect Transfer of Control of Licenses. The documents being provided in Attachment 2A and 3A contain proprietary financial information and financial projections related to the ownership and operation of the generation assets operated by ENO. These documents constitute proprietary commercial and financial information that should be held in confidence by the NRC pursuant to 10 CFR § 2.390(a)(4) because:

   i. This information is and has been held in confidence by ENO.

   ii. This information is of a type that is customarily held in confidence by ENO and there is a rational basis for doing so because the information contains sensitive financial information concerning projected revenues and operating expenses of ENO.

   iii. This information is being transmitted to the NRC voluntarily and in confidence.

   iv. This information is not available in public sources and could not be gathered readily from other publicly available information.

   v. Public disclosure of this information would create substantial harm to the competitive position of ENO by disclosing its internal financial projections.

3. Accordingly, ENO requests that the designated documents be withheld from public disclosure pursuant to 10 CFR § 2.390(a)(4).

Michael R. Kansler

STATE OF MISSISSIPPI)
)
COUNTY OF HINDS)

Subscribed and sworn to me, a Notary Public, in and for the State of Mississippi, this 30th day of July, 2007.

Amy A. Blaylock
Notary Public in and for the
State of Mississippi

Notary Public State of Mississippi At Large
My Commission Expires: June 17, 2009
Bonded Thru Helden, Brooks & Garland, Inc.

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Attachment 5

Form of SUPPORT AGREEMENT

Between

Entergy Nuclear Finance Holding, LLC

and

Entergy Nuclear Generation Company,
Entergy Nuclear FitzPatrick, LLC,
Entergy Nuclear Vermont Yankee, LLC,
Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Palisades, LLC

THIS SUPPORT AGREEMENT, dated as of _____ __, 2007 between Entergy Nuclear Finance Holding, LLC, a Delaware corporation ("Parent"), and Entergy Nuclear Generation Company, Entergy Nuclear FitzPatrick, LLC, Entergy Nuclear Vermont Yankee, LLC, Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Palisades, LLC (individually, "Subsidiary Licensee" and collectively, "Subsidiary Licensees"),

WITNESSETH:

WHEREAS, through its intermediate subsidiary companies, Parent is the indirect owner of 100% of the outstanding shares of the Subsidiary Licensees;

WHEREAS, the Subsidiary Licensees are the corporate entities that hold the NRC licenses for Pilgrim, Indian Point 2 and 3, FitzPatrick, Vermont Yankee, and Palisades (individually, each a "Facility," and collectively the "Facilities"); and

WHEREAS, Parent and the Subsidiary Licensees desire to take certain actions to assure the ability of the Subsidiary Licensees to pay the pro rata expenses of maintaining the Facilities safely and protecting the public health and safety (the "Operating Expenses") and to meet Nuclear Regulatory Commission ("NRC") requirements during the life of each Facility (the "NRC Requirements").

NOW, THEREFORE, in consideration of the mutual promises herein contained, the parties hereto agree as follows:

1. *Availability of Funding.* From time to time, upon request of Subsidiary Licensees, Parent shall provide or cause to be provided to Subsidiary Licensees such funds as the Subsidiary Licensees determine to be necessary to pay Operating Expenses and meet NRC Requirements; provided, however, in any event the aggregate unreimbursed amount which Parent is obligated to provide under this Agreement at any one time shall not exceed $700 million.

2. *No Guarantee.* This Support Agreement is not, and nothing herein contained, and no action taken pursuant hereto by Parent shall be construed as, or deemed to constitute, a direct or indirect guarantee by Parent to any person of the payment of the Operating Expenses or of any liability or obligation of any kind or character whatsoever of the Subsidiary Licensees. This Agreement may, however, be relied upon by the NRC in determining the financial qualifications of each Subsidiary Licensee to hold the operating license for a Facility.

3. *Waivers.* Parent hereby waives any failure or delay on the part of the Subsidiary Licensees in asserting or enforcing any of their rights or in making any claims or demands hereunder.

4. *Amendments and Termination.* This Agreement may not be amended or modified at any time without 30 days prior written notice to the NRC. This Agreement shall terminate at such time as Parent is no longer the direct or indirect owner of any of the shares or other ownership interests in a Subsidiary Licensee. This Agreement shall also terminate with respect to the Operating Expenses and NRC Requirements applicable to a Facility whenever such Facility permanently ceases commercial operations and certification is made as to the permanent removal of fuel from the reactor vessel.

5. *Successors.* This Agreement shall be binding upon the parties hereto and their respective successors and assigns.

6. *Third Parties.* Except as expressly provided in Sections 2 and 4 with respect to the NRC, this Agreement is not intended for the benefit of any person other than the parties hereto, and shall not confer or be deemed to confer upon any other such person any benefits, rights, or remedies hereunder.

7. *Other Financial Support Arrangements.* This Agreement supersedes any other support arrangement relating to NRC requirements, if any exists prior to the date hereof, between Parent or any other affiliate that is a signatory hereto, and a Subsidiary Licensee to provide funding when necessary to pay Operating

Expenses and meet NRC Requirements for the Facilities, and any such other financial support arrangement is hereby voided, revoked and rescinded. As such, the total available funding provided for in this Agreement shall be limited as set forth in Section 1 herein and shall not be cumulative with any other financial support arrangement for purposes of meeting NRC requirements that is subject to the jurisdiction of the NRC. For avoidance of doubt, the parties agree that this Section 7 does not apply to financial guarantees or commitments made to third parties, even where such agreements may relate to compliance with NRC requirements. A list of the financial support arrangements rescinded by this paragraph is provided as Schedule A.

8. *Governing Law.* This Agreement shall be governed by the laws of the State of Delaware.

IN WITNESS WHEREOF, the parties hereto have caused this Agreement to be executed and delivered by their respective officers thereunto duly authorized as of the day and year first above written.

ACKNOWLEDGED AND AGREED

Entergy Nuclear Finance Holding, LLC

By:
Name:_____
Title:_____

Entergy Corporation

By:
Name:_____
Title:_____

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Entergy International Holdings LTD

By:
Name:_____
Title:_____

Entergy International LTD LLC

By:
Name:_____
Title:_____

Entergy Nuclear Generation Company

By:
Name:_____
Title:_____

Entergy Nuclear FitzPatrick, LLC

By:
Name:_____
Title:_____

Entergy Nuclear Vermont Yankee, LLC

By:
Name:_____
Title:_____

Entergy Nuclear Indian Point 2, LLC

By:
Name:_____
Title:_____

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Entergy Nuclear Indian Point 3, LLC

By:
Name:_____
Title:_____

Entergy Nuclear Palisades, LLC

By:
Name:_____
Title:_____

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Schedule A

Guarantor	Guaranty on behalf of	Amount	Claim
Entergy International LTD LLC	Entergy Nuclear Generation Company	$50M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy International LTD LLC	Entergy Nuclear Indian Point 2, LLC	$35M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy International Holdings LLC	Entergy Nuclear Vermont Yankee, LLC	$35M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Corporation	Entergy Nuclear Vermont Yankee, LLC	$35M	If the financial assurance line is below $35M at the point that it is determined that ENVY will cease operations, ETR will make additional funds available to ENVY, up to $35M.
Entergy Corporation	Entergy Nuclear Vermont Yankee, LLC	$25M	If the financial assurance line is below $25M at the point that it is determined that ENVY will cease operations, ETR will make additional funds available to ENVY, up to $25M.
Entergy International LTD LLC	Entergy Nuclear FitzPatrick, LLC & Entergy Nuclear Indian Point 3, LLC	$50M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Global, LLC	Entergy Nuclear FitzPatrick, LLC	$20M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Global, LLC	Entergy Nuclear Indian Point 3, LLC	$20M	Guarantee to NRC for financial assurance to provide for safe plant operation and for decommissioning, through working credit line.
Entergy Global, LLC	Entergy Nuclear Palisades, LLC	$25M	Guarantee to NRC for financial assurance to provide for safe plant operation through working credit line.

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~~EXHIBIT I~~

Exhibit V

Synapse
Energy Economics, Inc.

FINANCIAL INSECURITY: The Increasing Use of Limited Liability Companies and Multi-Tiered Holding Companies to Own Nuclear Power Plants

Prepared by:
David Schlissel, Paul Peterson and Bruce Biewald
Synapse Energy Economics
22 Pearl Street, Cambridge, MA 02139
www.synapse-energy.com
617-661-3248

Prepared for:
STAR Foundation
Riverkeeper, Inc.

August 7, 2002

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List of Tables and Attachments

Foreword

Where Have All the Safeguards Gone

In nuclear power's first two decades, accident insurance requirements were seriously inadequate. Decommissioning costs were overlooked entirely. The 1979 accident at Three Mile Island undermined much nuclear complacency. In the early 1980s Congress and the U.S. Nuclear Regulatory Commission made serious efforts to address these shortcomings.

The nuclear self-insurance requirement – known as the Price-Anderson Act – was increased from $560 million to the current $9.3 billion, and each plant was required to set up a dedicated decommissioning trust fund to assure that funds would be available to clean up a closed plant.

With the passage of two more decades, renewed complacency has eroded these safeguards.

This report dissects a troublesome set of developments on the cusp between economic and safety regulation, namely the rearrangement of nuclear power plant ownership into the limited liability subsidiaries of a few large companies. Because this arrangement has occurred during an era of lax and dispirited regulation, some important issues have not been pursued effectively. As a result, the consolidation of nuclear ownership – although probably a positive development if carried out wisely – now risks the shifting of accident and decommissioning costs from the plant owners to the general public because the relatively secure financial backing of substantial utility companies has in many cases been replaced by a limited liability subsidiary whose only asset is an individual nuclear power plant.

With years of reckless undermining of economic and financial regulation now exposed in a series of catastrophic financial collapses, investigators turning over rocks keep finding the same agents of decay: demands for short term "performance" in the private sector compounded by regulatory cutbacks, underqualified commission appointments, Congressional hearings harassing public protection initiatives, pressure to deregulate more and faster-a ruinous mixture of money, pressure, overconfidence, complexity and ideology.

During all those years, health and safety regulation got the same debilitating treatment from Congress and the Presidency as its financial counterparts. How long before those chickens come home to roost, and where will the roosting be?

Even in the best of times, regulation tends to be reactive, responding to events or to applications. Rarely does a regulatory commission develop a set of affirmative requirements to guide those who seek its permits. Certainly neither the Nuclear Regulatory Commission nor the several economic regulators with jurisdiction over nuclear plants ever developed a comprehensive policy to guide those seeking to transfer nuclear plant ownership. Such a policy might have required a showing that the protection of the public was in no way diminished by these transfers. Or such a requirement might have been imposed as a condition of approving the transfers.

But it was not.

In the absence of any such requirement, public protection has depended on the acumen of a Nuclear Regulatory Commission unversed in financial matters and of economic regulators unversed in health and safety issues. As has happened in financial and in utility restructuring circles, fundamental safeguards have been circumvented.

Regulating in this way is like driving drunk. On any one occasion, there may be no consequences at all. But in the nuclear field the possibilities include the undermining of the scheme that assures compensation in the event of nuclear accidents and an increased likelihood that some of the costs of decommissioning nuclear power plants will be borne by the general public. Taxpayers, utility customers and powerplant neighbors who thought themselves protected by firm requirements may one day wear the stunned expressions of Enron retirement plan holders or WorldCom investors.

Clever advisors in several professions have no doubt been well rewarded for achieving these "deregulations." As they were at Global Crossing. As they were at Qwest. As they were at Andersen Consulting. But in the nuclear realm as in the others, they have been more clever than wise. The consequences remain to be revealed. We will be fortunate if the only harm is another blow to public confidence.

Peter Bradford[1]

[1] Visiting Lecturer in Energy Policy and Environmental Protection, Yale University; Former Chair, New York State Public Service Commission and Maine Public Utilities Commission; Former Commissioner, U.S. Nuclear Regulatory Commission; Past President, National Association of Regulatory Utility Commissioners.

Introduction

In recent years corporations have increasingly owned and operated nuclear power plants through multiple tiered holding companies, which frequently include limited liability companies ("LLCs"). LLCs are new organizational forms whose liability is limited to the specific assets they directly own. More than 25 nuclear power plants are today owned by such LLCs and additional corporate reorganizations can be expected. The use of complex organizational structures involving LLCs can shield the parent corporations and their shareholders from liabilities incurred by both direct and indirect subsidiaries. In so doing, the use of multi-tiered holding companies and LLCs to own and operate nuclear power plants raises several concerns regarding security, safety and potential federal and consumer liabilities.

Nuclear power plants were traditionally constructed and operated mainly by integrated investor-owned utilities under "cost-of-service regulation" through which necessary funds were provided to operate and decommission the plants safely. Starting in the mid-1990s, however, many nuclear power plant owners began to reorganize and to sell their nuclear units to unaffiliated companies or corporate affiliates. Some of these corporate reorganizations were required or encouraged as part of state efforts to deregulate the electric utility industry and to implement industry restructuring. Other reorganizations were adopted by plant owners, on their own initiative, in order to minimize tax liabilities, maximize flexibility in corporate ownership and management, and to protect corporate assets. According to the U.S. General Accounting Office ("GAO"), the U.S. Nuclear Regulatory Commission ("NRC") has reviewed more than 60 license transfer requests in recent years, affecting more than half of the nuclear plants in the nation.[2]

Synapse Energy Economics, Inc. ("Synapse") was asked by the STAR Foundation and Riverkeeper, Inc. to survey the increasing use of complex corporate ownership structures and LLCs to own and operate nuclear power plants and to review the NRC's oversight of these developments. Synapse also was asked to identify those areas in which changes need to be made to assure that there are adequate funds available to meet NRC-imposed requirements, including post September 11, 2001 security-related requirements and Price-Anderson Act nuclear accident insurance obligations and to assure that decommissioning funds are adequate and are protected. This Report presents our findings.

Data Sources

Synapse has used publicly available documents from the following sources in the preparation of this Report: the U.S. GAO, the U.S. NRC, corporate filings at the U.S. Securities and Exchange Commission, company websites, nuclear industry publications, utility filings at state regulatory commissions and answers to post-hearing questions that arose out of the January 23, 2002 Price-Anderson Act Hearings. The specific documents on which this Report is based are identified in footnotes or the list of references.

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[2] *Nuclear Regulation: NRC's Assurance of Decommissioning Funding During Utility Restructuring Could be Improved,* U.S. GAO Report, GAO-02-48, December 2001, at page 21.

This Report also relies on detailed publicly available information about the Entergy Corporation that Synapse obtained as a result of its work in Vermont Public Service Board Docket No. 6545 in which Entergy's proposed acquisition of the Vermont Yankee nuclear plant has been examined.

Conclusion

Over the last ten years, the ownership of an increasing number of nuclear power plants has been transferred to a relatively small number of very large corporations. These large corporations have adopted business structures that create separate limited liability subsidiaries for each nuclear plant, and in a number of instances, separate operating and ownership entities that provide additional liability buffers between the nuclear plant and its ultimate owners. The limited liability structures being utilized are effective mechanisms for transferring profits to the parent/owner while avoiding tax payments. They also provide a financial shield for the parent/owner if an accident, equipment failure, safety upgrade, or unusual maintenance need at one particular plant creates a large, unanticipated cost. The parent/owner can walk away, by declaring bankruptcy for that separate entity, without jeopardizing its other nuclear and non-nuclear investments. This report examines the recent trend towards the use of limited liability corporations in the nuclear industry, often as part of multi-tiered holding companies, and identifies numerous concerns related to the use of such business structures.

Summary of Findings

The above conclusion is based on the following findings:

Finding No. 1 - Nuclear power plant ownership and operation has become increasingly consolidated in a small number of very large corporations..

Finding No. 2 – Complex, holding companies, often including Limited Liability subsidiaries, are increasingly being used to own nuclear power plants.

Finding No. 3 – Limited Liability Companies are relatively new business structures that can enhance a parent corporation's ability to transfer funds from its subsidiaries and to shield assets from liability for financial risks.

Finding No. 4 –There continue to be significant financial and other risks associated with nuclear power plant ownership and operations.

Finding No. 5 – The NRC has expressed concern that deregulation can adversely affect the safety of operating nuclear power plants by increasing the pressure on licensees to reduce costs.

Finding No. 6 – The NRC has expressed concern that the use of holding company structures can reduce the assets that would be available for the safe operation and decommissioning of a nuclear power plant. However, the NRC does not adequately protect against the risk that an LLC subsidiary will transfer all of its operating profits to its parent company or engage in risky loans to or questionable deals with affiliated companies.

Finding No. 7 - The NRC's reviews of the financial qualifications of new nuclear power plant owners are inconsistent and may be too limited to ensure that subsidiaries will have adequate funds to safely operate and decommission their nuclear plants and pay retrospective Price-Anderson Act premiums.

Finding No. 8 – The financial guarantees that the NRC requires from prospective nuclear power plant owners may not be adequate to assure that plants are operated and decommissioned safely and that plant owners will be able to pay retrospective Price-Anderson Act insurance premiums in the event of a nuclear accident.

Finding No. 9 - The NRC has proposed to significantly reduce its review of a non-electric utility licensee's financial qualifications when it evaluates an application to renew a nuclear plant's operating license.

Finding No. 10 – The NRC does not require that parent corporations guarantee that funds will be provided to safely operate and decommission the nuclear power plants owned by their subsidiary companies.

Finding No. 11 – Taxpayers may be at risk if nuclear plant owning subsidiaries are unable to continue making safety-related or decommissioning expenditures or pay retrospective Price-Anderson Act premiums.

Finding No. 12 – 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.

Finding No. 13 – Case law indicates that it could be very difficult to hold a parent corporation responsible for the liabilities incurred by nuclear power plant-owning LLC subsidiaries in a multi-tiered holding company.

Finding No. 14 – The NRC has expressed serious doubts as to its ability to hold a parent corporation responsible for the liabilities incurred by a subsidiary.

Finding No. 15 – Shielding parent corporations from nuclear power plant operating, accident insurance, and decommissioning risks is unfair and economically inefficient.

Recommendations

1. Parent corporations should be required to guarantee that plant-owning subsidiaries and affiliates will be provided whatever funds are needed to safely operate and decommission their nuclear power plants.

2. Parent corporations should be held fully responsible for the unmet liabilities incurred by both direct and indirect nuclear power plant owning subsidiaries.

3. Congress should adopt legislation to assure that costs related to (1) safety and security (2) decommissioning assets and (3) Price-Anderson nuclear accident responsibilities receive priority in bankruptcy proceedings.

4. Reactor owners should be required to guarantee payment of their nuclear accident insurance responsibilities under the Price-Anderson Act through surety bonds,

letters of credit, sinking funds, or other comparable financial instruments that will be bankruptcy remote. This will assure that public liability claims will be paid up to the limits of the Price-Anderson Act without concern about the financial condition of the industry and without requiring a taxpayer bailout.

5. The Nuclear Regulatory Commission should not eliminate the current legal requirement that non-utility corporations must disclose their financial qualifications when applying to re-license nuclear power plants, as the agency has proposed in a recent rulemaking. Instead, the NRC should bolster its disclosure requirements concerning the character of the legal relationships between a parent corporation and its subsidiaries in the event of a bankruptcy, business failure or accident.

Finding No. 1 - Nuclear power plant ownership and operation has become increasingly consolidated in a small number of very large corporations.

In the past, a relatively large number of utilities around the nation owned nuclear power plants or, at least, were joint owners with other companies. However, as a result of industry restructuring, nuclear power plant ownership has become increasingly consolidated in a small number of large corporations. In fact, as shown in Table No. 1 below, ten corporations currently own all or part of 70 of the 103 nuclear power plants in the U.S.

Table No. 1
Concentration of Nuclear Power Plant Ownership

Parent Corporation	Number of Operating Nuclear Units Owned (in whole or in part)
Exelon Corporation	19
Entergy Corporation	10[3]
Duke Energy	6
Dominion Resources, Inc.	6
Southern Company	6
TVA	5
Progress Energy	5
FPL Group	5[4]

[3] Includes the Vermont Yankee Nuclear Station.

[4] Includes the Seabrook Nuclear Station.

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- Commonwealth Edison experienced numerous simultaneous extended outages among the eight units at its Dresden, LaSalle, Quad Cities, and Zion nuclear stations. For example, during the first six months of 1996, the utility had at least three units shut down at any one time for extended outages of longer than three months in duration. Commonwealth Edison had at least four units shut down at any one time for extended outages during the last six months of 1996, except for a short period at the end of August and early September. The utility also experienced simultaneous outages of at least six months in length at its two unit Zion nuclear station from October 1993 through April 1994 and at its two unit LaSalle Station from September 1996 through 1998.
- Both units at the D.C. Cook Nuclear Plant in Michigan were shutdown from September 1997 through June 2000.
- Both units at the Salem Nuclear Station were shutdown for more than two years between July 1995 and August 1997.
- Both units at the Brunswick nuclear plant were shutdown for the twelve month period April 1992 through April 1993.
- Both units at the Calvert Cliffs nuclear plant were shut down at the same time for more than one year starting in May 1989.

Finding No. 2 - Complex, multi-tiered holding companies, often including limited liability subsidiaries, are increasingly being used to own nuclear power plants.

Except for those power plants owned by municipal utilities and the Yankee Nuclear Plants in the Northeast, nuclear units historically were directly owned by integrated investor-owned utility companies which owned other generating facilities and had significant transmission and distribution assets as well. Over the past five to ten years, however, corporations have established multiple tiered holding companies through which they indirectly own nuclear power plants. Except for the Exelon Corporation, these new nuclear power plant owning subsidiaries generally own only a single asset, i.e., an individual nuclear power plant, or both units at a multiple unit site.[6]

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[6] The nuclear industry's interest in single asset nuclear generating companies is not new. It dates back to the 1960s, perhaps even to the 1950s, when the plans were developed for the ownership of the Yankee Rowe and Connecticut Yankee nuclear plants. Then, in the late 1980s and early 1990s, some companies, including Middle South Utilities (subsequently renamed "Entergy") and General Public Utilities, reorganized, creating specific corporate entities to operate ***but not own*** their nuclear power plants. In one notable case in Michigan, however, the Consumers Power Company proposed transferring a poorly performing nuclear plant, Palisades, to a new corporate entity, PGCo, created for the sole purpose of owning and operating the plant. This ill-conceived proposal was designed to shift nuclear-related risks away from the Company, placing them instead upon consumers and the public. For more information, see Bruce Biewald, "Do We Really Need Nuclear Generating Companies?," in Public Utilities Fortnightly, June 7, 1990. and the Direct Testimony of Bruce Biewald, submitted on behalf of the Attorney General of Michigan, April 19, 1989 in Michigan Public Service Commission Case No. U-9172.

The corporate subsidiaries included in these complex ownership chains are increasingly chartered as Limited Liability Companies ("LLCs"). As we will discuss in Finding No. 3 below, LLCs are relatively new business structures that enhance a parent corporation's ability both to transfer funds from its nuclear-power plant owning subsidiaries and to shield its other assets from liability from the financial risks associated with its nuclear operations.

The following examples illustrate the accelerating trend in the nuclear industry to use multiple tiered holding companies and LLC subsidiaries to own and operate nuclear plants. It is important to note that each of the parent corporations listed in these examples also has numerous other subsidiaries unrelated to its nuclear power plant ownership.

Exelon Corporation

Exelon Corporation was formed in 2000 by the merger of Unicom (Commonwealth Edison Company's parent) and PECO Energy Company. Commonwealth Edison's 10 operating nuclear plants have been transferred to Exelon Generation Company, LLC, ("EGC") which is a wholly owned subsidiary of Exelon Ventures Company, LLC, which, in turn, is a wholly-owned subsidiary of Exelon Corporation. PECo's Limerick and Peach Bottom nuclear plants also have been transferred to EGC, as has PECo's ownership interest in the two Salem Nuclear Plants.

PECo also owned 50 percent of the AmerGen Energy Company, LLC, ("AmerGen") which had acquired and operated three nuclear power plants in the U.S.: Three Mile Island Unit 1, Clinton, and Oyster Creek. PECo's interests in AmerGen have been transferred to EGC, LLC. Consequently, through EGC, LLC, Exelon Corporation owns and operates part or all of 16 nuclear plants and owns part of another three units.

The current organizational structure through which Exelon owns these nuclear assets is illustrated in Attachment No. 1 to this Report.

Entergy

Entergy Corporation was a pioneer in establishing separate corporate entities to own and operate nuclear power plants. Entergy today owns and operates ten nuclear units through an extensive network of wholly-owned subsidiaries.

Entergy currently owns five nuclear units in the South through five wholly-owned retail public utility companies and another wholly-owned subsidiary, System Energy Resources, Inc.[7]

Entergy also has purchased another five nuclear units in the Northeast including its just completed purchase of the Vermont Yankee nuclear plant. As shown in Attachment No. 2 to this Report, Entergy owns each of these units through a multi-tiered series of subsidiaries, many of which are limited liability companies. For example, the Indian

[7] Entergy Arkansas, Inc., Entergy Gulf States, Inc., Entergy Louisiana, Inc., Entergy Mississippi, Inc., and Entergy New Orleans, Inc.

Constellation Energy Group, Inc.	4
FirstEnergy	4

The six largest owners alone own part or all of 52 nuclear units, or one-half of all of the operating nuclear power plants in the nation.

At the same time, the Nuclear Management Company ("NMC") holds the NRC-issued operating licenses for eight nuclear plants in the Midwest. However, each of the utilities involved in NMC continues to own its own plants, is entitled to the energy generated by the plants, and retains the financial obligations for the plants safe operation, maintenance and decommissioning.

This industry consolidation may yield significant benefits in terms of economics, safety and reliability. However, it also raises the possibility that simultaneous extended outages of more than one nuclear plant will leave a "fleet" owner without needed revenues to fund safety-related expenses or capital expenditures at its other facilities. At the same time, the increasing consolidation of ownership also raises the possibility that an owner will have to bear Price-Anderson Act retrospective burdens measured in the hundreds, not tens, of millions of dollars, possibly without adequate revenues from which to make such payments.[5]

In fact, there have been numerous instances where two or more of a company's nuclear plants have been out of service at the same time for six months or longer due to problems that arose as a result of an emphasis on reducing costs, deficiencies in the utility's safety culture, management problems, or generic or plant-specific technical issues. For example:

- Two of the three units at the Palo Verde Nuclear Generating Station were shut down at the same time for approximately twelve months starting in March 1989. During this same twelve month period, the third Palo Verde unit was shut down for numerous outages, including one outage that lasted approximately four months.
- The two units at the South Texas nuclear plant were both shut down for the twelve month period February 1993 to February 1994.
- All five of TVA's operating nuclear power plants were shut down in 1985. The first unit to be restarted, Sequoyah Unit 1, re-commenced commercial operations in May 1989.
- Northeast Utilities' Millstone Units 2 and 3 were shut down for multi-year outages between March 1996 and June 1998. Millstone Unit 1 was shutdown in November 1995 and permanently retired in 1997.

[5] In the event of an accident at one of the nation's nuclear power plants, the Price-Anderson Act requires nuclear plant owners to make "retrospective" (i.e., post-accident) payments after the initial $200 million tier of insurance is exhausted. Under the Act's present terms, and given the number of operating plants, this obligation is a maximum of $88.085 million per unit with a maximum of $10 million per year. Consequently, an owner of multiple units could face retrospective obligations of hundreds of millions of dollars in total and tens of millions per year.

Point 2, Indian Point 3, and Fitzpatrick nuclear units are each owned by a separate LLC.[8] In the case of Indian Point 2, the immediate owner is Entergy Nuclear IP2, LLC. This company 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.[9]

The structure through which Entergy owns the Indian Point 3 and Fitzpatrick units is even more complex because each of the LLCs that owns these plants is, in turn, 50 percent owned by two other indirect Entergy subsidiaries, Entergy Nuclear New York Investment Company I and Entergy Nuclear New York Investment Company II. As shown in Attachment No. 2, these two Entergy Nuclear New York Investment Companies are themselves subsidiaries of Entergy Nuclear Holding Company #1 which, in turn, is a wholly-owned subsidiary of Entergy Corporation.

Another Entergy subsidiary, Entergy Nuclear Operations, Inc. ("ENO") operates Entergy's nuclear units in the Northeast.[10] Additional services are provided by other Entergy subsidiaries such as Entergy Services, Inc. (management, administrative and support services) and Entergy Nuclear Fuels Company (nuclear fuel planning, procurement and related services).

Entergy has provided the following explanation for this tiered holding company structure:

> Entergy Nuclear Holding Company, a first tier of Entergy Corporation, has been established with the intent that it will ultimately hold all the subsidiaries associated with Entergy's nuclear operations. This will consolidate all of Entergy's unregulated nuclear operations under a single holding company, while still supporting the operational and financing demands of the individual plants. *The use of holding companies below Entergy Nuclear Holding Company allows Entergy to segregate various types of financing, investment and business activities, and by doing so, enables Entergy to better manage and control risks associated with these activities.*[11] (Emphasis added)

Remarkably, Entergy has indicated that only two of all of the subsidiaries included in Attachment 2 -- ENO and Entergy Nuclear Generation Company, which owns and operates the Pilgrim Nuclear Station -- have any employees other than officers.[12] The

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[8] Although the wholly-owned subsidiary that currently owns Entergy's Pilgrim Station is not an LLC, Entergy has said that it will seek to change the form of that subsidiary to an LLC in the near future.

[9] Entergy has said that ultimately all of subsidiaries associated with Entergy's nuclear operations will be owned by Entergy Nuclear Holding Company. Rebuttal Testimony of Connie Wells, Entergy Nuclear Vermont Yankee, LLC, in Vermont Public Service Board Docket No. 6545, at page 9.

[10] Entergy's nuclear units in the South are operated by yet another subsidiary, Entergy Operations, Inc.

[11] Rebuttal Testimony of Entergy Nuclear Vermont Yankee witness Connie Wells in Vermont Public Service Board Docket No. 6545, dated February 25, 2002, at page 9.

[12] Entergy response to Department of Public Service Information Request No. 2-10 in Vermont Public Service Board Docket No. 6545.

rest of the listed subsidiaries are merely paper organizations. In addition, the subsidiaries listed on Attachment 2 share many of the same individuals as officers.[13]

The NRC requires licensees of deregulated nuclear plants to provide certain financial guarantees that a unit would have sufficient funding to enable the licensee to continue to maintain the unit in a safe manner in case of an extended outage or a premature shutdown. Entergy's financial guarantees for its deregulated units in the Northeast are provided by two subsidiaries not listed in Attachment 2 -- Entergy International Holdings, LTD LLC and Entergy Global Investments, Inc. Both of these subsidiaries are themselves holding companies.[14]

As shown in Attachment No. 3, Entergy also has a very extensive network of other subsidiaries, in addition to those that own and operate its deregulated nuclear units in the Northeast.

Dominion

Dominion Resources, Inc. ("DRI") owns the two operating nuclear power plants at Millstone Point in Connecticut through a multi-tiered chain of subsidiaries. As shown on Attachment No. 4, DRI owns Dominion Energy Holdings, Inc. which, in turn, owns Dominion Energy Inc., LLC which owns Dominion Nuclear, Inc.. Dominion Nuclear, Inc. then owns Dominion Nuclear Marketing I, Inc, Dominion Marketing II, Inc, and Dominion Marketing III, LLC that together own Dominion Nuclear Connecticut, the direct owner of the Millstone nuclear station.[15]

Dominion also owns the four nuclear units at its North Anna and Surry stations in Virginia through the Dominion Generation Corporation which is a wholly-owned subsidiary of Dominion Energy Holdings, Inc. Dominion Generation Corporation also will own the fossil and hydro facilities that were formerly owned by Virginia Power Company.

Constellation

Constellation Energy Group, Inc. ("Constellation") purchased 100 percent of the Nine Mile Point Unit No. 1 nuclear plant and 82 percent of Nine Mile Point Unit No. 2 nuclear plant in 2001. Both of these units are located in upstate New York, near the City of Oswego. When Constellation sought NRC approval to transfer the units' licenses it also requested approval to complete a complex fourteen step corporate realignment. The nuclear-related results of this proposed realignment are shown on Attachment No. 5. The direct owner of the two Nine Mile Point nuclear plants is Nine Mile Point Nuclear Station, LLC, which is a wholly owned subsidiary of Constellation Nuclear Power Plants,

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[13] Synapse has learned greater detail about Entergy's current holding company structure through its involvement on behalf o the Vermont Department of Public Service in Vermont Public Service Board Docket No. 6545.

[14] Entergy response to Department of Public Service Information Request No. 1-42(c) in Vermont Public Service Board Docket No. 6545.

[15] Dominion's August 17, 2001 letter to the NRC concerning the Millstone Nuclear Power Station Corporate Restructuring.

Inc, which, in turn, is a wholly owned subsidiary of Constellation Nuclear, LLC. Constellation's other two nuclear plants are owned by another subsidiary of Constellation Nuclear Power Plants, Inc, Calvert Cliffs Nuclear Power Plant, LLC. Constellation also has numerous other nuclear-related subsidiaries. Constellation Nuclear, LLC is, in turn, a subsidiary of Constellation Energy Group, Inc.

The parent corporation resulting from this corporate realignment will be BGE Corporation which will own Constellation Energy Group, Inc., as an immediate subsidiary.

Other Companies

The owners of fleets of nuclear power plants are not the only corporations that have established multi-tiered holding companies to own their nuclear plants. For example, as part of its proposed reorganization to recover from bankruptcy, Pacific Gas & Electric is seeking permission to transfer its two Diablo Canyon Nuclear Plants to a new LLC subsidiary, Diablo Canyon LLC. As shown on Attachment No. 6, this subsidiary would, in turn be a wholly owned subsidiary of Electric Generation, LLC, which in turn is a subsidiary of the Newco Energy Corporation, a wholly owned subsidiary of PG&E Corporation.[16]

Another example is Public Service Enterprise Group ("PSEG") which owns and operates the Salem and Hope Creek nuclear plants and is part owner of the Peach Bottom Nuclear generation station through a line of wholly-owned subsidiaries that includes PSEG Power, LLC, and its wholly-owned subsidiary, PSEG Nuclear.

Finding No. 3 - Limited Liability Companies are relatively new business structures that are used to shield the assets of a parent corporation from liability for financial risks.

The fundamental purpose and rationale for the creation of a "corporation" is to allow investors to pool their resources to engage in a business activity while limiting the financial consequences or "liability" of each individual investor. The most typical arrangement is for an investor to purchase stock or "shares" in the corporation. The money or other value paid for the shares is the limit of that investor's personal liability. The corporation's total liability is limited to the value of its investors' shares, plus any insurance policies that may be applicable.

Partnerships, an alternative form of business organization, are characterized by the inability of the partners to limit their individual liability. Each partner is wholly and personally responsible for all debts of the business. This onerous feature of partnerships has led to the development of many variations on the partnership model, particularly the limited partnership, as a way to shield some or all of the partners from unlimited liability.

Looking only at the liability issue, one might wonder why partnerships are ever chosen as a business structure. There are two primary reasons: streamlined management and lower

[16] PG&E's November 30, 2001 Application to the NRC for License Transfers and Conforming Administrative License Amendments.

taxes. A corporation is required to have articles of incorporation, a board of directors, and a management structure separate from the board of directors. Partnerships can be much more flexible with the same individual, or group of individuals, performing both day-to-day management and decision-making functions. Tax policy significantly favors partnerships by allowing all business profits to flow directly to the partners where they are taxed on their business income along with any other personal income. Corporations, because they are considered a separate entity, must pay corporate taxes before profits can flow to its investors, who then pay taxes on their corporate income on an individual basis. This is commonly called the "double taxation" feature of corporations.[17]

The dilemma facing entrepreneurs who want to start a business is whether the business structure should be designed to protect their existing personal assets by limiting their liability (a corporation) or whether the business structure should be designed to allow them to maximize their income from this single venture through lower taxes (a partnership). As discussed below, the nuclear industry seeks to achieve both liability protection and maximum income through the use of new limited liability corporate structures.

Limited Liability Subsidiaries

Limited liability companies (LLCs) are relatively new business structures that combine features of corporations and partnerships. An LLC has the same limited liability of corporations, but has the management flexibility of a partnership. Most significantly, pursuant to an IRS ruling in 1988, an LLC is considered a partnership for federal income tax purposes.

The first LLCs in the United States were formed in Wyoming in 1977 for foreign corporations that wanted to invest in very risky mineral exploration and development. Since 1977, LLC statutes have been enacted in all fifty states. They have proven to be a particularly attractive business structure for investments in high-risk ventures. LLCs can be formed by individuals, partnerships, or corporations. They can be managed by the LLC members (owners) or by an elected group of members, or by a single member. The management choice also acts to specify the members who can legally bind the LLC through contracts with outside entities.[18]

LLCs have become a very attractive business structure for corporations that acquire nuclear power plants. By creating a separate LLC for each nuclear plant, the profits from each plant's operations can flow back to the parent corporation without any intervening tax liability. The parent corporation's liability for each plant is limited to the investment the parent corporation made in initially setting up the LLC. Also, there can be more than one LLC between the parent corporation and the most risky component of the overall

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[17] For general background on business structures, see any of a number of law school texts on business organizations. The information above is derived from "Organizing Limited Liability Companies: The Trend Continues", Richard M. Fijolek, Practicing Law Institute (1997); "A Limited Liability Company Checklist", Jerome P. Friedlander, II, Federal Lawyer (March/April 1995); and "The ABCs of LLCs, Steven Auderieth, Vermont Bar Journal and Law Digest (February, 1995).

[18] See above, Friedlander and Auderieth.

investment. For example, the technical support services for several nuclear plants can be consolidated into a separate LLC that contracts with all the individual plant LLCs. If one nuclear plant becomes unprofitable and goes into bankruptcy proceedings, in theory, only the single plant LLC assets are in jeopardy; the technical services LLC can continue to provide services to all the other single plant LLCs.

A particular concern regarding the use of LLCs is the situation where a parent corporation inserts several layers of LLCs between itself and the entity operating a high-risk business. Each of those intervening LLCs can act as a barrier to extending liability to the parent corporation that contains most of the assets. As noted in the case studies in Finding No. 2 of this Report, this approach appears to have been embraced by the parent corporations that recently have been purchasing nuclear plants. If a nuclear plant was unable to cover its liabilities, it might require several separate litigations, or a very large and complex single litigation, to pierce all the corporate veils back to the parent corporation with the bulk of the assets.

Finding No. 4 –There continue to be significant financial and other risks associated with nuclear power plant ownership and operations.

The restructuring of electricity markets has meant increased risks for owners of any deregulated electric generation facilities, whether their plants are fossil-fired or nuclear. Revenues which used to be based on traditional "cost of service" concepts and stable rates are now based instead on the actual sales from a power plant at market prices that are sometimes volatile.

At the same time, there are significant nuclear-related risks that could have a material adverse effect on nuclear power plant owners. For example, a recent Prospectus issued by Exelon Corporation for the sale of $700 million of notes by Exelon Generation Company, LLC specifically identified the following risks associated with owning and operating nuclear power plants:

> We may incur substantial cost and liabilities due to our ownership and operation of nuclear facilities. The ownership and operation of nuclear facilities involve certain risks. These risks include: mechanical or structural problems; inadequacy or lapses in maintenance protocols; the impairment of reactor operation and safety systems due to human error; the costs of storage, handling and disposal of nuclear materials; limitations on the amounts and types of insurance coverage commercially available; and uncertainties with respect to the technological and financial aspects of decommissioning nuclear facilities at the end of their useful lives. The following are among the more significant of these risks:
>
> Operational risk. Operations at any nuclear generation plant could degrade to the point where we have to shut down the plant. If this were to happen, the process of identifying and correcting the causes of the operational downgrade to return the plant to operation could require significant time and expense, resulting in both lost revenue and increased fuel and purchased power expense to meet our supply commitments. For plants operated by us but not wholly

owned by us, we could also incur liability to the co-owners. We may choose to close a plant rather than incur substantial costs to restart the plant.

Regulatory risk. The NRC may modify, suspend or revoke licenses and impose civil penalties for failure to comply with the Atomic Energy Act, the regulations under it or the terms of the licenses of nuclear facilities. Changes in regulations by the NRC that require a substantial increase in capital expenditures or that result in increased operating or decommissioning costs could adversely affect our results of operations or financial condition.

Nuclear accident risk. Although the safety record of nuclear reactors generally has been very good, accidents and other unforeseen problems have occurred both in the United States and elsewhere. The consequences of an accident can be severe and include loss of life and property damage. Any resulting liability from a nuclear accident could exceed our resources, including insurance coverages.

These same risks apply to other nuclear plants including those owned and operated by multi-tiered holding companies and LLCs.

The industry's expressed desire to build new nuclear plants also can be expected to increase the financial pressures on licensees as they may have to further reduce O&M expenditures at existing plants in order to fund the construction of new ones.

Finding No. 5 - The NRC has expressed concern that deregulation can adversely affect the safety of operating nuclear power plants by increasing the pressure on licensees to reduce costs.

Although it has been said that an efficient and economical plant is often a safe plant,[19] the NRC has expressed concern that the transition to economic deregulation can adversely affect nuclear power plant safety and may not provide the same degree of assurance that adequate funds would be provided for safe operation and decommissioning.[20]

The NRC has further explained the impact that increased competition can have on nuclear power plant economics and safety:

> As described in SECY-97-253, traditional "cost-of-service" regulation, under which virtually all NRC power reactor licensees have operated, has typically been effective in providing necessary funds for licensees to operate and decommission their nuclear plants safely. With the advent of greater competition within the electric utility industry, pressures to reduce costs and improve efficiency have increased and will almost certainly intensify as deregulation proceeds. Moreover, with deregulation of the generation sector of the industry, traditional cost-of-service regulation is likely to essentially disappear for most generators. Thus, the concept of electric utility, as

[19] *NRC Staff Requirements Memorandum, SECY-98-153*, dated June 29, 1998, at page 3.

[20] *NRC Final Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry* (62 *Fed. Reg.* 44071; August 19, 1997)

currently defined in 10 CFR 50.2 may in the future no longer be meaningful for a large number of, if not all, power reactor licensees. Electricity rates set by competition in a free market may not provide the same degree of assurance of adequate funds for safe operation and decommissioning as traditional cost-of-service ratemaking. In SECY-97-253, the staff cited the example of the "Independent Safety Assessment of Maine Yankee Atomic Power Company" (NRC Staff Report: Ellis W. Merschoff, Team Lead; October 1996), which concluded, "Economic pressure to be a low-cost energy producer has limited available resources to address corrective actions and some plant improvement upgrades.

When the NRC issued its Final Policy Statement on the Restructuring and Economic Deregulation of the Electric Utility Industry (62 *Fed. Reg.* 44071; August 19, 1997), specific safety concerns with respect to rate deregulation and restructuring were identified. For example, the final policy statement discussed such safety concerns as reductions in expenditures for manpower and training and other reductions in operations and maintenance (O&M) and capital additions budgets. The issues of on-line maintenance and increased fuel burnup were also addressed.

In addition, with respect to specific plants such as Maine Yankee, Millstone, and others, the inspection process has identified several manifestations of inappropriate responses to competitive pressures. These include: increased need for corrective actions; maintenance operator work-arounds; temporary modification and procedure revision backlogs; decreased performance in operator licensing and requalification programs; increased frequency of significant operational and occupational safety events; decreased plant and system reliability; increased volume and acrimony of allegations; and increased frequency of regulatory violations and resulting penalties.

As deregulation proceeds, cost pressures may increase these types of reductions in safety margins at plants. Moreover, because the impact of budgetary reductions can cut across all plant safety-related programs, other impacts in addition to those previously identified may occur as a result of deregulation. For example a merchant plant with no assets other than the nuclear plant itself could be unable to make necessary safety expenditures after an extended outage if it did not have an adequate financial cushion to pay costs incurred during the outage. In such a situation, it is not clear that a transition from indefinite shutdown to permanent shutdown and decommissioning would be sufficiently smooth to prevent funding shortages from causing safety problems during the shutdown transition period. That is, given the requirements in 10 CFR 50.82 with respect to: (1) the limitation on the use of the trust fund for legitimate decommissioning activities; and (2) the timing of significant decommissioning trust fund withdrawals, a licensee could run out of funds for operational safety expenses before it was able to draw on its decommissioning trust fund. This, in turn, could force the NRC to

make the decision for the licensee to permanently cease operations and initiate decommissioning pursuant to 10 CFR 50.82.[21]

The nuclear industry itself has acknowledged the safety and economic risks associated with economic deregulation. For example, a former President of the industry's American Nuclear Society told the Society's Winter 2001 meeting that "Safety is the highest priority because of the impact on cost that would result from an NRC-forced shutdown" and that there is now "actually a higher focus on safety than before."[22] However, he also noted the challenges that come from deregulation and restructuring:

> With restructuring comes challenges for plant operators and regulators, Quinn continued. These challenges for operators include management focus on economics, not safety; pressure on workers to keep plants operating (because of volatility of electricity prices); pressure to reduce preventative maintenance; deferral of equipment replacements; and less investment for safety backfits. For the regulator, these include increased workload (because of mergers, license transfers, etc.); pressure to avoid requiring shutdowns of plants; and increased political pressure to reduce the regulatory burden. Challenged also is the nuclear technology infrastructure. According to Quinn, there is less cooperation among competing nuclear utilities, and less safety research and technical support for the plants.[23]

Finding No. 6 - The NRC has expressed concern that the use of holding company structures can reduce the assets that would be available for the safe operation and decommissioning of a nuclear power plant. However, the NRC does not adequately protect against the risk that an LLC subsidiary will transfer all of its operating profits to its parent company or engage in risky loans to or questionable deals with affiliates.

The NRC Staff has expressed concern that the use of holding company structures can lead to a diminution of the assets necessary for the safe operation and decommissioning of a licensee's nuclear power plant.[24] In fact, as early as March 1993 the NRC Staff expressed concern that:

> Current and potential organizational structures of many power reactor licensees and their corporate affiliates are complex and evolving. The staff believes that the public health and safety implications of such structures warrant further examination. A licensee subsidiary without assets other than

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[21] *NRC Staff Requirements Memorandum, SECY-98-153*, dated June 29, 1998, at pages 2 and 3.

[22] *ANS Winter Meetings: Nuclear Power - Attracting Notice, A Brighter Outlook*, Nuclear News, August 2001, starting at page 34.

[23] Ibid.

[24] *Safety Evaluation by the NRC's Office of Nuclear Reactor Regulation "Related to Proposed Corporate Restructuring of Commonwealth Edison Company,"* October 5, 2000, at page 3.

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> the licensed reactor could renege on its decommissioning obligations if forced to shut down prematurely. Given that corporate law generally limits the liability of stockholders, the NRC may not have recourse to the assets of a parent company if its subsidiary defaults absent legally enforceable commitments by owners. Case law with respect to bankruptcy proceedings is also ambiguous. Although bankruptcy courts have generally directed bankruptcy trustees to make justifiable, legally required expenditures to protect public health and safety, it is not clear that these expenditures will always have a high priority relative to other claims. The staff believes that it should evaluate possible ways to increase assurance of decommissioning funds availability. An increased degree of confidence may be appropriate to assure that the problems that the Office of Nuclear Material Safety and Safeguards has had with some of its licensees abandoning materials sites prior to cleanup will not be experienced for power reactor licensees.[25]

The NRC Staff consequently requested that the NRC Commissioners approve publication of an advance notice of proposed rulemaking to explore alternatives to mitigate the potential impact on safety of power reactor licensee ownership arrangements and to consider whether increase assurance of funding availability for decommissioning activities was needed.

> *A licensee subsidiary without assets other than the licensed reactor could renege on its decommissioning obligations if forced to shut down prematurely.*
>
> NRC Staff, March 1993

Unfortunately, the NRC Commissioners disapproved this request and, instead, asked for additional information on the staff proposal. In response to a Commission question on how many reactor licensees could try to set up a corporate veil to avoid decommissioning costs, the NRC Staff noted:

> Potentially, any investor-owned utility could establish a holding company to which it could transfer the bulk of its assets over time. If forced to shut down prematurely, a licensee with assets limited essentially to the shut down reactor could declare bankruptcy and renege on any unfunded decommissioning obligation. If a bankrupt licensee had insufficient assets, a bankruptcy court might be powerless to order that assets of a parent company be used to fund decommissioning, even if the court wished to do so.[26]

In the years since 1994, the NRC has not developed or adopted any policy limiting the transfer of operating profits from the subsidiary that directly owns a nuclear plant. Nor

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[25] *Issuance of An Advance Notice of Proposed Rulemaking on the Potential Impact on Safety of Power Reactor Licensee Ownership Arrangements, SECY-93-075,* March 24, 1993, at page 1.

[26] *Response to Staff Requirements Memorandum of April 28, 1993, Which Disapproved Issuance of An Advance Notice of Proposed Rulemaking on the Potential Impact on Safety of Power Reactor Licensee Ownership Arrangements, SECY-94-280,* at pages 4 and 5

does the NRC have any policy limiting the types or magnitudes of the loans that such an operating subsidiary can make to affiliated companies.

At most, the NRC merely conditions license transfer approvals to new holding company structures upon a requirement that the licensee not transfer to its proposed parent or any other affiliated company significant assets for the production, transmission or distribution of electric energy without first notifying the NRC. The NRC has defined "significant assets" to be facilities having a "depreciated book value exceeding 10% of the company's consolidated net utility plant."[27]

The NRC also does not have a specific policy statement or procedure on how limited liability companies or other types of licensees use financial assurance funds in the forms of lines of credit for plant operation.[28] Nor does the NRC have any specific policy statement or procedure that controls how it would consider approval of requests of limited liability companies to reduce, replace, or withdraw available lines of credit that are subject to NRC conditions. Instead, the NRC has said that it will review such requests on a case-by-case basis.[29]

The NRC has explained its policy for addressing situations where a licensee has drawn upon the lines of credit provided by a parent or affiliated companies. In such situations, the NRC would:

> evaluate the reasons behind [the licensee's] drawing on the lines of credit. The staff cannot provide a detailed discussion of potential agency actions until it learns the specific reasons for the usage of such funds. Generally, if drawings on the lines of credit were made to cover short-term cash flow deficiencies that did not appear to have any significant safety ramifications, the NRC would not likely need to take any specific action. If drawing on the lines of credit were to indicate serious longer-term financial problems that appeared to potentially adversely impact protection of public health and safety, the NRC would monitor the effects of any degradation on protection of public health and safety and act appropriately.[30]

The NRC's failure to have any policy limiting the transfer of operating profits from the subsidiary that directly owns a nuclear plant or the types or magnitudes of the loans that such an operating subsidiary can make to affiliated companies is all the more significant because the new holding companies also may have not set policies governing these matters. For example, Entergy has said that there are no written procedures governing the distribution of operating profits from the subsidiaries that are the direct owners of its

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[27] For example, see the October 5, 2000 Safety Evaluation by the NRC Office of Nuclear Reactor Regulation of the proposed corporate restructuring of PECO Energy Company, at page 3.

[28] Enclosure 1 to the NRC's December 13, 2001 letter to Christine Salembier, Commissioner, Vermont Department of Public Service, on the subject of "Vermont Yankee Nuclear Power Station – Lines of Credit Associated with Vermont Yankee License Transfer."

[29] Ibid.

[30] Ibid.

nuclear units.[31] These subsidiaries either make distributions to their immediate parent companies or make loans to affiliated companies depending on the specific cash requirements of the parent companies or the affiliates.

Vermont Department of Public Service witness Andrea Crane has explained to the Vermont Public Service Board why it should be concerned about the ability of a parent corporation to drain the funds available to a nuclear power plant-owning subsidiary:

> ...in addition to being concerned about the availability of capital for ENVY's[32] operations, there is also a concern that Entergy Corp. may threaten the long-term financial viability of ENVY by using ENVY's earnings to fund other Entergy Corp. operations, leaving insufficient funds in ENVY for nuclear operations. Therefore, in addition to raising concerns about the availability of sufficient operating and capital funds, I am also concerned about the need to retain capital in ENVY. The Board should avoid a repeat of the situation that transpired in PG and E ... whereby funds were transferred from a successful operating entity to the holding company, leaving the operating company in dire financial straits.[33]

Ms. Crane also expressed concern about the absence of formal Entergy corporate policies governing the transfer of profits and inter-affiliate transactions:

> The lack of direct control over its internally generated funds, and the vagueness of the corporate policy, does not provide an appropriate level of financial assurance for the ownership and operation of a nuclear power plant. It leaves open the possibility that Entergy Corp could require 100% of operating earnings as dividends from its subsidiaries, including ENVY, if it needed funds to meet other priorities or emergencies, leaving the owners of the nuclear plants without sufficient capital to pursue their own immediate priorities.[34]

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[31] Entergy Response to Department of Public Service Information Request No. 2-36 in Vermont Public Service Board Docket No. 6545.

[32] ENVY is Entergy Nuclear Vermont Yankee LLC, which is the Entergy Corporation subsidiary that will own the Vermont Yankee nuclear plant if the purchase is approved by the Vermont Public Service Board.

[33] Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 9.

[34] Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 28.

Finding No. 7 - The NRC's reviews of the financial qualifications of new nuclear power plant owners are inconsistent and may be too limited to ensure that subsidiaries will have adequate funds to safely operate and decommission their nuclear plants and pay retrospective Price-Anderson Act premiums.

Before it allows a nuclear power plant operating license to be transferred, the NRC conducts reviews of the financial qualifications of the prospective owner. The NRC's regulations specify the types of information that a prospective licensee must provide and the nature of the review that must be conducted by the NRC staff.

However, the applicable NRC regulation, 10 CFR 50.33(f), is inconsistent in that on the one hand it says that "the applicant shall submit information that demonstrates the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the **period of the license**." (emphasis added) But the regulation then merely requires applicants to submit estimates for total annual operating costs for only the first 5 years of operation of the facility. Although the NRC can ask for information for subsequent years, this regulation can mean that the NRC will only review five years of operating cost data when the new owner may be seeking transfer of a license which will continue in effect for another 25 years or longer.

In reviewing the financial qualifications of a prospective licensee, the NRC requires that the new owner either meet a supply and demand test or show that it has an investment grade rating or equivalent from at least two bond-rating organizations. The supply and demand test examines whether the prospective licensee will earn sufficient revenues (either from the sale of electric power from the nuclear plant or from other sources) to cover expected operational expenses at the plant.[35] This analysis is based on the applicant's uncertain and speculative estimates of total operating revenues and costs for the first full five years following the expected completion of the license transfer.[36] At the same time, it is very unlikely that the new corporate subsidiaries that actually will own the transferred plant will have issued any securities that had received investment grade or equivalent ratings from any bond-rating organizations.

If a prospective licensee is unable to meet either the supply and demand test and or the bond rating criteria test, the NRC will consider its ability to fund a six-month outage. Although assuring the funding for a six-month outage is not required where a prospective licensee meets either of the NRC's two primary tests, in those cases where a prospective licensee voluntarily guarantees the funds to pay for a six-month outage, the Commission will accept that commitment and impose a licensee condition prohibiting the applicant from voiding or diminishing those guarantees.

The U.S. General Accounting Office ("GAO") has evaluated the NRC's review of the financial qualifications of prospective licensees to safely operate and decommission

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[35] NUREG-1577, Revision 1, at Section III.1.b.

[36] It also appears that the NRC does not consider the need to pay retrospective Price-Anderson Act premiums when it considers a prospective licensee's financial qualifications to safely operate and decommission a nuclear power plant.

nuclear power plants. The GAO concluded that for the most part, the NRC's reviews of new owners financial qualifications have enhanced the level of assurance that they will safely own and operate their plants in a deregulated environment and not need to shut them down prematurely.[37]

However, the GAO also found that the NRC did not always adequately verify the new owners' financial qualifications.[38] In particular, the GAO concluded that when the NRC reviewed the financial qualifications of Exelon to safely own and operate the largest fleet of nuclear plants in the U.S., it did not require the same additional guarantees from the parent or affiliated companies that the new owner would have sufficient revenues to cover the plants' operating costs as it had required from other proposed license transfers.[39] The NRC also did not validate the information submitted by the new owner to demonstrate that the company was financially qualified.[40] In fact, the GAO concluded that the NRC had eventually transferred the licensees to Exelon Generation Company on the basis of projected financial information that both the affected companies and the NRC knew to be inaccurate.[41]

The NRC's review of financial qualifications continues after a license is transferred. Each licensee is required to submit an annual financial report, pursuant to 10 CFR 50.71(b) and a decommissioning funding status report is required every two years.[42] The NRC Staff also monitors the general financial status of nuclear plant licensees by screening the trade and financial press reports, and other sources of information.[43]

However, it is unclear whether the NRC has the staff resources or the expertise to conduct adequate reviews of licensee's financial qualifications. For example, the NRC's Executive Director for Operations informed the Commissioners in April 1997 that the expertise of the NRC Staff in matters of finance and economic analysis were "limited."[44] At the same time, the size of the NRC Staff has been reduced by approximately ten percent since 1997.[45]

The NRC has expressed confidence in its Staff's ability to identify financial distress and has quoted approvingly a Staff member who said "severe financial distress from any of the licensees is something that's not going to be hidden from view very long."[46] However, the suddenness of ENRON's collapse and the apparent absence of public warnings of that

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[37] *Nuclear Regulation: NRC's Assurances of Decommissioning Funding During Utility Restructuring Could be Improved,* GAO-02-48, December 2001, at page 6.

[38] Ibid., at page 4.

[39] Ibid., at page 21.

[40] Ibid., at pages 21 and 31-32.

[41] Ibid., at page 33.

[42] 10 CFR50.75(f)(1).

[43] NUREG-1577, Rev 1, Section III.1.d., at page 5.

[44] NRC SECY-97-071, April 2, 1997.

[45] NUREG-1350, Vol. 13, Figure 4.

[46] *In the Matter of Power Authority of the State of New York and Energy Nuclear Fitzpatrick,* 53 N.R.C. 488, June 21, 2001.

company's severe financial distress prior to that collapse suggest that the NRC may not have any warning about a licensee's impending financial problems.

Finally, the NRC recently has indicated its intention to reduce the regulatory burden on licensees by eliminating the requirement that licensees include financial qualifications information in license renewal applications.[47] This would mean that there would be no assessment of the financial qualifications of a licensee to safely operate a nuclear power plant for up to an additional twenty years beyond the expiration of its existing NRC-issued license.

In conclusion, there are a number of reasons to have serious concerns about the quality of the NRC's review of the financial qualifications of licensees and prospective licensees.

Finding No. 8 - The financial guarantees that the NRC requires from prospective nuclear power plant owners may not be adequate to assure that plants are operated and decommissioned safely and that plant owners will be able to pay deferred Price-Anderson Act insurance premiums in the event of a nuclear accident.

The NRC has generally accepted guarantees from prospective nuclear power plant licensees in the range of $55 to $75 million to pay for a six-month outage. However, in a number of cases the licensee has not offered and the NRC has not required the licensee to make any such guarantee.[48] For example, there appears to be guarantees in place for only three of the nuclear units owned by Exelon Generation Company, LLC. These are the three units that were originally 50 percent owned by PECO Energy Company and were transferred to Exelon Generation Company, LLC as part of the merger between Unicom and PECO Energy. The guarantees that were in place when the plants were owned by PECO Energy and British Energy were transferred along with the plants. However, it does not appear that there is any guarantee in place for the other 16 nuclear plants that are currently owned by Exelon Generation, Company, LLC.

There is no evidence that these limited $55 to $75 million guarantees will provide sufficient funds to enable power plant owners to safely shutdown their nuclear plants in case of a serious event or significant problem and to maintain the plant in a safe shutdown condition until the problem is addressed or the licensee is able to gain access to the plant's decommissioning trust fund. For example, a substantial number of nuclear power plants have been shutdown since January 1996 for outages that lasted far longer than six months:

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[47] FedNet Government News, June 5, 2002.

[48] As we will discuss in Finding No. 9, Constellation has guaranteed that its nuclear power plant-owning subsidiaries, Nine Mile Point LLC and Calvert Cliffs Nuclear Plant LLC will receive whatever cash is needed to protect the public health and safety.

Table No. 2
Nuclear Power Plant Outages
Since June 1995
That Lasted Nine Months or Longer

Plant	Period Shutdown	Outage Duration
Beaver Valley 2	December 1997 - September 1998	9 months
Clinton	September 1996 - May 1999	32 months
Cook Unit 1	September 1997 - December 2000	39 months
Cook Unit 2	September 1997 - June 2000	33 months
Indian Point 2	February 2000 - December 2000	10 months
Kewaunee	September 1996 - June 1997	9 months
LaSalle Unit 1	September 1996 - August 1998	23 months
LaSalle Unit 2	September 1996 - April 1999	31 months
Millstone Unit 2	February 1996 - May 1999	39 months
Millstone Unit 3	March 1996 - June 1998	27 months
Point Beach Unit 1	February 1997 - December 1997	10 months
Point Beach Unit 2	October 1996 - August 1997	10 months
Salem Unit 1	May 1995 - April 1998	35 months
Salem Unit 2	June 1995 - August 1997	26 months

Indeed, as Table No. 1 (in Finding No. 1 above) and Table No. 2 reveal, it is not unusual for more than one unit at a single site to be shutdown for an extended outage at the same time. These simultaneous extended outages could significantly increase the financial pressures on the units' owner in a deregulated environment when its cash flow depends on the actual sales from the plant rather than on regulated rates for an entire utility.

Moreover, it is not unreasonable to expect that a nuclear unit might be shutdown for more than six months before the ultimate parent corporation makes the decision to permanently retire the unit. After all, the full extent of the plant's problems and the expense and time it would take to repair and restart the unit might not be apparent until the plant had been shut down for a substantial period of time.

This could mean that all of the funds guaranteed by an affiliate or the parent corporation could be exhausted before the licensee would be able to gain access to the unit's decommissioning fund. For example, Millstone Unit 1 was shutdown for 31 months before Northeast Utilities decided in July 1998 to permanently retire the plant. Commonwealth Edison Company's Zion Units 1 and 2 were shutdown for eleven and sixteen months, respectively, before the Company decided in January 1998 to

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permanently retire both plants. The Maine Yankee plant was shut down for eight months before its Board of Directors decided in August 1997 to permanently retire it.

But even if an outage were shorter than six months, the maintenance and/or capital expenditures required to repair a plant and restore it to service may be significantly higher than the company had projected in its application to the NRC. The limited funds pledged by a parent corporation or an affiliate could be inadequate under such circumstances.

Finding No. 9 – The NRC has proposed to significantly reduce its review of a non-electric utility licensee's financial qualifications when it evaluates an application to renew a nuclear plant's operating license.

The NRC has proposed to eliminate the requirement that non-electric utility power reactor licensees submit financial qualifications information in their license renewal applications.[49] At the same time, the NRC also has proposed to require the submission of such information when utilities reorganize and operate as "non utility" generators.

The NRC's proposal to require financial reviews when a utility recognizes with a new financial structure is important. However, the decision to reduce disclosure obligations on nuclear power plant owners when they seek renewal of operating licenses for up to 20 years creates the potential for added risk of non-performance in critical areas.

A formal and rigorous review at the time of license renewal for aging nuclear reactors is a particularly appropriate time to evaluate the financial requirements. It is at this point that a business plan can be evaluated over the proposed lifetime of a licensee's facility. The financial resources needed to address the safe and secure operations, make capital improvements to a complex 30 year old machine, meet added license conditions required after the events of September 11, 2001, and to meet decommissioning and public liability obligations under the Price Anderson Act, must be juxtaposed against the economic conditions in the electricity markets and the availability of capital and insurance.[50]

The NRC's justification for not requiring a financial qualifications review at the time of relicensing is that it can monitor licensees when changes take place in licensee's financial qualifications. These day-to-day or limited annual reviews are not substitutes for a

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[49] June 4, 2002 Federal Register, at Vol. 67, No. 104, pp. 38427.

[50] The wisdom of looking into the future was underscored in the case of USEC, Inc, which has an NRC Certificate under 10 CFR Part 76 to operate uranium enrichment plants. The NRC conducted a financial review of the USEC, Inc. Certificate when it was issued in 1998 using the threshold of a current investment grade credit rating. The NRC determined USEC was reliable and economic based on its BBB+ investment grade debt rating. However, the NRC did not look beyond the 5 year term of the certificate to evaluate USEC's financial qualifications or the company's ability to operate with an unsustainable business model. If it had, it could have readily foreseen that USEC's financial condition would deteriorate over time due to a number of factors including the declining value of its sales contract, lower market prices, increasing unit costs of output and lack of competitive technology to enrich uranium for nuclear power reactors. These factors led to multiple credit downgrades and subsequent NRC doubts about whether USEC's economic resources were sufficient to be recertified for another 5 years.

formal, rigorous and disciplined review examining all a licensee's financial ability to fulfill its obligations for safely and securely operating an aging reactor in a competitive marketplace.

Historically, the ratemaking process for a utility corporation had provided reasonable assurance that a license applicant would have funds necessary to operate a reactor. In these circumstances, a licensee could be assumed of obtaining all of the reasonable funds it needed to continue operating its aging power plant. However, non-utility generators now lack the same assured funding, and as utilities diversify into telecommunications, trading operations and high-risk financial activities, the risk that there will be insufficient capital grows. To provide a green light for 20 years of operation without a rigorous review of a licensee's financial resources and business plans invites unwelcome surprises.

Finding No. 10 - The NRC does not require that parent corporations guarantee that funds will be provided to safely operate and decommission the nuclear power plants owned by their subsidiary companies.

The NRC does not require that a parent corporation guarantee the funds that may be needed to operate and decommission safely the nuclear power plants owned by subsidiaries. Instead, the NRC Staff has included conditions requiring a parent guarantee in the orders approving license transfers as additional assurance of financial qualifications only when such a guarantee has been offered by the applicant.[51].

For example, in its reviews of the financial qualifications of Entergy Corporation to own the Pilgrim, Indian Point 2, Indian Point 3, Fitzpatrick, and Vermont Yankee nuclear plants, the NRC has accepted guarantees that would be provided through lines of credit from affiliated financial subsidiaries which may not have sufficient liquid capital when it is needed by a plant-owning affiliate. One of these credit lines is to be used for working capital, if needed. The other is not intended to be used in the normal course of business but instead would be used in the event of problems at the plant. Entergy has indicated that this line of credit would be used to pay the costs between the unplanned shutdown of a plant and the availability of funds from the plant's decommissioning trust fund.[52]

Vermont Department of Public Service witness Andrea Crane has explained the problems that can arise from the fact that neither of the Entergy subsidiaries that provide these lines of credit have any physical assets:

> The result is that these two companies are only as strong as 1) their receivables from, and investment in, associated companies, and 2) Entergy Corp's commitment to provide them with additional funds, if required. Entergy Corp, therefore, has full discretion as to whether or not to provide sufficient capital to EIHL and EGI so that these two financing vehicles can

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[51] *In the Matter of GPU Nuclear, Inc and AmerGen Energy Company, LLC*, 51 N.R.C. 193, at Footnote No. 8.

[52] Prefiled Direct Testimony of Michael R. Kansler, Entergy Nuclear Vermont Yankee, Vermont Public Service Board Docket No. 6545, at page 10.

> meet their commitments to ENVY. If Entergy Corp should choose to walk away from EIHL and EGI, there appears to be no recourse for ENVY.[53]

For this reason, Ms. Crane recommended that the parent Entergy Corporation be required to guarantee that the pledged funds actually would be available if needed:

> Entergy Corporation should be obligated to stand behind the total financial exposure occasioned by the ownership and operation of this nuclear power plant. It is not reasonable to allow Entergy Corporation to shield itself from financial responsibility with complex financial arrangements. It certainly should not be allowed to offer guarantees from subsidiaries that do not have sufficient assets to meet their obligations on a stand-alone basis, because the parent could walk away from those subsidiaries if its own interests so dictated. If Entergy Corporation intends to stand behind the guarantees of its subsidiaries, it should have no problem in making the guarantee directly.[54]

Even though the NRC had accepted the $70 million guarantee provided by the two lines of credit from Entergy Corporation subsidiaries, in response to the concerns raised by the Ms. Crane and the Vermont Public Service Board, the parent Entergy Corporation has provided an additional financial guarantee of up to $60 million.[55] As Entergy has explained:

> The intent of that guarantee is to make sure that, in the event of a premature shutdown of the Vermont Yankee Nuclear Power Station, there will be money available to bridge the gap between shutdown and the point at which ENVY is able to access the decommissioning trust fund. Thus, if either line of credit has been drawn upon, Entergy will guarantee to make up any deficiency up to a total of $60 million.[56]

Entergy also acknowledged that the parent corporation has not provided a similar guarantee in support of **any** of the other nuclear plants its subsidiaries have acquired.[57] It further noted that state and federal regulators, including the NRC, had found the smaller guarantees by affiliated companies, not the parent corporation, to be sufficient.[58]

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[53] Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 18.

[54] Direct Testimony of Andrea Crane on behalf of the Vermont Department of Public Service, Vermont Public Service Board Docket No. 6545, at page 22.

[55] Ms. Crane subsequently testified that the revised commitments by the parent Entergy Corporation adequately addressed the concerns in her Direct Testimony. Supplemental Testimony of Andrea Crane in Support of the Memorandum of Understanding in Docket No. 6545, at page 2 of 9.

[56] Rebuttal Testimony of Connie Wells, Entergy Nuclear Vermont Yankee, LLC, in Vermont Public Service Board Docket No. 6545, at page 3, lines 8-13.

[57] Ibid., at page 5, lines 1-5.

[58] Ibid.

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Dominion has voluntarily committed $150 million from the parent corporation, DRI, to assure that Dominion Nuclear Connecticut (the new owner of the Millstone Nuclear Station) will have sufficient funds available for meeting its operating expenses for the recently acquired Millstone Units 2 and 3.[59] Dominion has explained that the subsidiary, Dominion Nuclear Connecticut, has the right to obtain such needed funds from DRI as it determines "are necessary to protect the public health and safety, meet NRC requirements, meeting ongoing operational expenses or to maintain Units 2 and 3 safely."[60] However, it does not appear that Dominion has made the same commitment to the four nuclear plants at the Surry and North Anna sites owned by the Dominion Generation Corporation.

Constellation has guaranteed that each of its nuclear power plant-owning subsidiaries, i.e., Nine Mile Point Nuclear Station LLC and Calvert Cliffs Nuclear Plant LLC, would be provided whatever cash is needed to protect the public health and safety.[61]

But it does not appear that the parent Exelon Corporation has guaranteed any funds to its power plant owning and operating subsidiary Exelon Generation Company, LLC.

Finding No. 11 – Taxpayers may be at risk if nuclear plant owning subsidiaries are unable to continue making safety-related or decommissioning expenditures or pay retrospective Price-Anderson Act premiums.

In attempting to assure the Vermont Public Service Board that the former owners of the Vermont Yankee nuclear plant and their ratepayers are unlikely to be required to pay any shortfalls in decommissioning funds, Entergy has noted that the NRC has on several occasions said that the burden of paying any such shortfalls would fall on taxpayers:

> NRC regulations do not specifically address the potential liability of other parties in the event that the licensed owner is unable to provide the funds required for decommissioning. In the past, the NRC indicated that any failure of the licensed owner to meet its decommissioning funding obligations would result in a burden on taxpayers -- presumably in the form of a publicly funded cleanup. See, e.g., SECY-94-280 (Nov. 18, 1984), at 4. ("Such action would either increase the potential risk to public health and safety of the decommissioning process or would shift the burden of decommissioning funding from ratepayers to taxpayers.") (emphasis added); 61 Fed. Reg. 15427, 15428 (Apr. 8, 1996)("The liability of the licensee to provide funding for decommissioning may adversely affect protection of the public health and safety. Also, a lack of decommissioning funds is a financial risk to taxpayers

[59] *Dominion August 31, 2000 Application for the transfer of the licenses for Millstone Units 1, 2 and 3*, at page 10.

[60] Ibid.

[61] *Calvert Cliffs Nuclear Power Plant Request for a Transfer in Control*, December 20, 2000, at page 9 and *Nine Mile Point Unit Nos. 1 & 2 NRC License Transfer Application*, February 1, 2001, at page 23.

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(i.e., if the licensee cannot pay for decommissioning, taxpayers would ultimately pay the bill. (emphasis added)."[62]

In fact, there are a number of possible circumstances in which taxpayers could be asked to bear much, if not all, of the cost of a major power plant accident. First, there is no assurance that the primary tier of insurance would be available to a licensee in the event of an act of terrorism against a nuclear power plant. American Nuclear Insurers has testified that it would only have resources available to provide the primary insurance coverage to cover a single act of terrorism.[63] Thereafter, all licensees would be left without any primary insurance coverage. At that point, licensees might seek recourse in the courts for a finding that domestic terrorism is an "act of war." Acts of war are excluded from coverage under the Price-Anderson Act.[64]

At the same time, the liabilities associated with a nuclear accident are borne by every nuclear power plant owner in the U.S. as a result of the pooling of liabilities for accidents with claims in excess of $200 million. The maximum cost per reactor is $88.085 million (subject to inflation adjustments) in secondary liability. As shown on Table No. 3 below, the liability for nuclear owners with multiple plants, such as Exelon (19 units) and Entergy (10 units), could approach or exceed $1 billion.

Table No. 3
Potential Price Anderson Act Nuclear Insurance Liabilities

Parent Corporation	Maximum Potential Annual Liability	Maximum Potential Total Liability
Exelon Corporation	$163.52 million	$1,440.35 million
Entergy Corporation[65]	$99 million	$872.04 million
Duke Energy	$52.50 million	$462.45 million
Dominion Resources, Inc.	$57.03 million	$502.32 million
Southern Company	$39.16 million	$344.94 million

[62] Legal Memorandum on the "Decommissioning Liability Associated with a Power Reactor License," Goodwin Procter LLP, February 24, 2002, submitted by Entergy Corporation to the Vermont Public Service Board as Exhibit ENVY-Wells-3 to the Prefiled Rebuttal Testimony of Connie Wells in Docket No. 6545.

[63] John Quattocchi, Senior Vice-President, American Nuclear Insurers, February 15, 2002 Response to Question from Senator Reid, Hearing before the Senate Committee on Environment and Public Works, January 23, 2002.

[64] The NRC's "opinion" is that claims arising out of an act of terrorism at a nuclear power plant would not be excluded under the Price Anderson Act. February 13, 2002 NRC Answer to Question No. 3 from Senator Reid, Hearing before the Senate Committee on Environment and Public Works, January 23, 2002. However, the NRC recognizes that a "question of this nature and magnitude" would likely need to be resolved by a court in the first instance.

[65] Potential Liability figures reflect Entergy ownership of the Vermont Yankee Nuclear Station.

TVA	$60 million	$528.51 million
Progress Energy	$44.72 million	$393.87 million
FPL Group[66]	$47.33 million	$416.91 million
Constellation Energy Group, Inc.	$38.20 million	$336.49 million
FirstEnergy	$40 million	$352.34 million

However, under the Atomic Energy Act, a licensee's secondary liability can be deferred if it would constitute an undue hardship on the licensee.[67] In such a situation, the secondary liability that would have been borne by the license would become a taxpayer funded liability. It is not unreasonable to expect that power plant owners, especially those that are thinly capitalized, will try to avail themselves of this deferral should a major accident occur.

Moreover, this Report has focused on nuclear-related issues. Nuclear power plants also contain large amounts of asbestos and large volumes of toxic chemicals. Taxpayers also could be forced to bear the costs of cleaning up for these and any other non-nuclear-related pollutants if a single asset power plant-owning subsidiary was able to successfully declare bankruptcy and a court was unwilling to hold the parent corporation liable.

Finding No. 12 - 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.

NRC regulations require any nuclear power plant licensee to immediately report any filing of a voluntary or involuntary petition for bankruptcy.[68] However, the NRC has no additional financial requirements for situations where a licensee files for bankruptcy or otherwise encounters financial difficulties. Nor does the NRC have any statutory authority to require a licensee which is in bankruptcy to continue to make safety-related or decommissioning payments or to pay retrospective Price-Anderson Act premiums. The NRC must intervene in the proceedings before the bankruptcy court and petition the court to require such payments.

The NRC has acknowledged that the license transfer requirements contained in 10 CFR 50.80 do not specifically or expressly refer to a prospective licensee's ability to meet financial protection payments that may be required under the Price-Anderson Act.[69] However, the NRC has said that 10 CFR 140.21 requires reactor licensees that are covered under the Price-Anderson system to provide annual guarantees of payments of

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[66] Potential Liability figures reflect FPL Group ownership of the Seabrook Nuclear Station.

[67] Atomic Energy Act Section 170(b)(2)(A) and (2)(B).

[68] 10 CFR 50.54 (cc).

[69] NRC February 13, 2002, response to Post-Hearing Question 6 from Senator Reid.

retrospective premiums and that the NRC evaluates an applicants guarantees of payment of retrospective premiums when it considers a license transfer request. [70]

The NRC has further said that it annually reviews the Price-Anderson Act guarantees for all of its power reactor licensees, including those that are LLCs.[71] All of the licensees have, to date, used the cash flow method of guarantee allowed under 10 CFR 140.21; that is, a licensee may demonstrate that it has sufficient cash flow over 3 months to meet an annual $10 million retrospective premium payment for each reactor that it owns.[72] As long as the licensee chooses that method and is able to pass the financial test for cash flow each year, no additional guarantee is required. However, if a licensee were not able to pass the cash flow test, it would have to provide some other allowable guarantee such as surety bonds, letters of credit, revolving credit/term load arrangements, maintenance of escrow deposits of government securities, or such other type of guarantee as might be approved by the NRC.[73] But there is no requirement that the parent corporation provide such a guarantee, only the subsidiary, and there is no requirement that resources be available to pay the maximum of $88.085 million per reactor.

The NRC has stated that under 10 CFR 140, a licensee is required to pay the retrospective premium, notwithstanding its financial status.[74] The NRC also has said that its has had positive experiences with bankruptcy courts that have overseen the Chapter 11 reorganizations of Public Service Company of New Hampshire (Seabrook nuclear plant), Cajun Electric Cooperative (River Bend), El Paso Electric Company (Palo Verde), and Vermont Electric Generation & Transmission Cooperative (Millstone 3).[75] According to the NRC, in each of these cases, the bankruptcy courts allowed these bankrupt licensees to pay all safety-related operational and decommissioning expenses (including, the NRC believes, Price-Anderson primary layer and on-site property insurance premium payments). The NRC also has noted that during its bankruptcy PG&E has continued to meet all safety-related expenses for its nuclear plants.

However, the NRC has acknowledged that it could potentially face a conflict with other claims in a bankruptcy proceeding "if there were an accident sufficient to trigger a retrospective premium assessment. The NRC would presumably require a licensee to pay the assessment, but the bankruptcy court could order the licensee not to pay it."[76]

In addition, the NRC's earlier experience with the bankruptcies all involved entities that owned a number of different assets. The bankruptcy of a single-asset LLC, which owns only a single nuclear power plant, would present very different circumstances and

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[70] NRC February 13, 2002, response to Post-Hearing Question 6 from Senator Reid.

[71] The NRC also requires that each licensee submit an annual financial report, 10 CFR 50.71(b) and a decommissioning fund status report every two years (and annually during the last five years of operation). 10 CFR 50.71(f)(1).

[72] A retrospective premium is insurance that is paid after an accident.

[73] NRC February 13, 2002, response to Post-Hearing Question 8 from Senator Reid.

[74] NRC February 13, 2002, response to Post-Hearing Question 9 from Senator Reid.

[75] NRC February 13, 2002, response to Post-Hearing Question 2 from Senator Inhofe.

[76] NRC February 13, 2002, response to Post-Hearing Question 9 from Senator Reid.

challenges. At the same time, as we will discuss later in this Report, given the multi-tiered holding companies (including LLCs) through which parent corporations now own many nuclear power plants, the NRC might have trouble "piercing the corporate veil" to require a parent of a bankrupt LLC subsidiary to make the required retrospective premium payments.

It is clear that there are no specific statutory or regulatory safeguards in place to ensure that retrospective premiums under the Price-Anderson Act will be available from bankrupt nuclear plant-owning subsidiaries or from their parent corporations. The NRC has sought legislation from Congress to ensure that decommissioning costs receive explicit priority in bankruptcy proceedings. But, so far, that legislation has not been enacted.[77] The NRC has further stated its willingness to support legislation to prioritize safety-related claims in bankruptcy proceedings and to avoid any potential conflict between NRC requirements to pay into the retrospective Price-Anderson Act premium pool and other claims in bankruptcy.[78]

Finding No. 13 – Case law suggests that it would be very difficult to hold a parent corporation responsible for the liabilities incurred by nuclear power plant owning LLC subsidiaries in a multi-tiered holding company.

As mentioned earlier in this Report, the multiple layers of subsidiaries, including LLCs, that have been created by parent corporations in the nuclear industry are a cause of serious concern. Even if a court concludes that the liability of the subsidiary that actually operates the nuclear plant should be extended to business structures above it (for example, if under capitalization and profit distributions have left the subsidiary unable to cover the costs of unanticipated repairs or security improvements and the subsidiary decides to cease operations), the ability of the court to find a senior business entity with sufficient capital could be complicated by multiple layers of subsidiaries and LLCs. There may be issues of jurisdiction, applicable state or federal statutes, the role of the NRC, and other myriad issues of law and fact that would need to be resolved. Given that the presumption in every state and federal statute is for the limitation of corporate liability, the burden is always on the party trying to extend that liability to show that the law, facts, and public policy all support violating the statutory presumption.[79] Courts, in

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[77] The Energy Policy Act of 2002 (HR 4), as approved by the U.S. Senate, amends the U.S. Bankruptcy Code to prevent creditors in a bankruptcy proceeding from attaching an NRC licensee's decommissioning funds until the decommissioning has been completed. The Senate enacted provision also seeks to prevent creditors from using Price-Anderson insurance and those deferred premiums held in reserve to satisfy creditors. However, neither version of the Energy Policy Act of 2002, that enacted by the House or the Senate, would require a parent corporation or other guarantor to commit resources in the event that there are not adequate resources within a bankrupt LLC to satisfy claims after a nuclear accident. Post accident liabilities could shift to taxpayers in this case.

[78] NRC February 13, 2002, response to Post-Hearing Question 9 from Senator Reid.

[79] "Piercing the Corporate Veil: An Empirical Study", Robert B. Thompson, 76 Cornell Law Review 1036 (1991), Section II, and "Limited Liability and the Corporation", Frank H. Easterbrook and Daniel R. Fishel, 52 U. Chi. L. Rev. 89 (1985), Section IV.

general, are reluctant to pierce the corporate veil and extend liability; when multiple corporations are involved, that reluctance only increases.

Despite the limitations on corporate liability embodied in statutes, there are numerous instances where courts have been willing to ignore those limitations under a wide range of factual circumstances. The case law varies a great deal from one state to another, but all of them involve some rationale for "piercing the corporate veil" and holding the owners of the corporation personally liable. For the purposes of this Report, it is important to note that in the nuclear power industry, the owners of a nuclear power plant-owning LLC subsidiary are most likely to be another LLC or a parent corporation. The objective of the effort to pierce the corporate veil in this situation would be to make the parent corporation responsible for the liability of the LLC subsidiary.

There is an enormous volume of litigation over the issue of extending liability through to the owners of a corporation. The case law is varied and complex and a thorough and complete review is beyond the scope of this project. What follows is a summary of the common themes that have been used by a variety of courts for extending liability.[80]

Starting from a presumption that a corporation's liability is limited, facts must be presented to justify extending liability. Some of the fact situations that have been persuasive to courts are the following:

- Corporate form is used as a front for illegal or fraudulent activity. In these cases, courts express no reluctance in holding individuals liable for the debts of the corporation since there is no public policy that seeks to support such activities under any business structure.
- Corporate form is used as a sham or a mere shell to avoid liability. In these cases, the individuals or parent corporation are aware from the start that the corporation is unlikely to ever repay its debts or liabilities and seek to acquire as much income as possible before creditors foreclose.
- Individual owners subvert the corporation for their personal gain. In these cases, the personal enrichment may or may not be based on illegal or unethical actions. If the facts establish that owners personally benefited from corporate activities (beyond the normal sharing of corporate profits), then courts are generally willing to make them personally liable. These cases often involve members of the Board of Directors or managers. Corporate owners who do not personally benefit but are aware of the enrichment of other owners can be held personally liable based on their breach of their fiduciary responsibilities to the corporation.
- Under-capitalization of the corporation. In these cases, there is a determination that at the time of incorporation, or due to subsequent management actions, there is insufficient capital available for the business activities of the corporation. Although similar to the problem of a sham corporation, the decision by the court

[80] Id., Thompson at 1063-1072; Easterbrook at 109-113.

involves a more objective analysis of appropriate levels of capitalization for similar entities engaged in similar activities.

- Improper distributions of income. These cases involve decisions by the corporate management, usually the Board of Directors, to distribute corporate income to shareholders in a financially irresponsible manner that leaves the corporation unable to meet its obligations. These are very fact-specific litigations that involve a great deal of hindsight analysis. However, if the facts show a clear pattern of irresponsibility, as opposed to poor business decisions, courts will extend liability to specific individuals or the corporation in general.

- Interference in management. These cases involve situations where owners, often large stockholders in closely held corporations, become so involved in corporate management that they look more like a managing partner than just an investor. Courts will extend liability to these "investors" on the theory that they do not deserve the normal corporate protection.

- Environmental, regulatory, or public policy. These factors are often included with one or more of the above fact patterns to support extending liability. It is unusual for a court to invoke "public policy" by itself as a justification for piercing the corporate veil.

An empirical study of court decisions where piercing the veil issues were litigated indicates that courts are very reluctant to impute liability to the shareholders of public corporations. Closely-held corporations (non-public and usually with few investors) and related corporate entities (subsidiaries, affiliates, etc.) are the forms to which courts have applied extended liability.[81]

There is very little case law involving LLCs that specifically addresses piercing the corporate veil due to the relatively short time period (fifteen years) during which LLC structures have been developed. Consequently, there is great uncertainty as to the effect that having one or more LLCs in the ownership chain within a holding company will have on the willingness of a court to pierce the corporate veil in order to hold a parent company responsible for the liabilities of its indirect nuclear power plant owning-subsidiary.

Finding No. 14 - The NRC has expressed serious doubts as to its ability to hold a parent corporation responsible for the liabilities incurred by a subsidiary.

There are two NRC cases that involved attempts to pierce the corporate veil of the operator of a nuclear power plant. In 1995, the NRC attempted to negate a transfer of assets from a licensee which, as part of a complicated corporate restructuring, had become a subsidiary to a newly created holding company because the transfer had occurred without the prior written consent of the NRC, as required by section 184 of the Atomic Energy Act. The NRC held that it could pierce the veil of corporations that

[81] Id., Thompson at 1070.

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violate section 184. However, before a final adjudication, this case ended in a settlement.[82]

In 1997, the NRC tried to force a parent company to provide additional funds to the decommissioning fund for a subsidiary plant. However, prior to a final adjudication, the NRC approved a settlement that resolved the decommissioning fund issue without any specific finding as to the parent company's liability.[83] In accepting the settlement, the NRC expressed concern that there was a "substantial possibility of defeat if the case proceeds to trial [on a theory of] piercing the corporate veil."

Both cases were cited in a legal memorandum provided by the current owners of the Vermont Yankee Nuclear Power Corporation, which concluded that attempts to pierce the corporate veil of nuclear power plant subsidiaries were unlikely to succeed and have seldom been attempted.[84] Despite the numerous specific instances where courts have extended liability to parent corporations, there is great uncertainty as to whether or not courts would apply such extended liability to multi-layered nuclear power companies.

Finding No. 15 – Shielding parent corporations from nuclear power plant operating and decommissioning risks is unfair and economically inefficient.

To the extent that the organizational structures discussed above serve to successfully shield the parent company from risks, they are inequitable and undermine efficient decision-making.

As a matter of fairness, individuals and companies should take responsibility for cleaning up after themselves. If an unanticipated problem in operation causes a nuclear plant to experience an extended or permanent outage prior to the end of its operating license or if the decommissioning of a plant turns out to cost more than expected, then the parent company may decide to provide additional resources to the subsidiary in order to carry out the subsidiaries responsibilities. On the other hand, the parent company may not. If there are clean up costs which the subsidiary is unwilling to bear, then these may fall upon taxpayers. Considerations of fairness would have the company that profited (or expected to profit) from plant operation bear the costs of cleaning up the facility.

This is also a matter of economic efficiency. If a company is protected from significant risks associated with its decisions, then there is what economists call an "externality." A reasonable definition of externality is provided in a popular economics textbook as follows:

> An externality or spillover effect occurs when production or consumption inflicts involuntary costs or benefits on others; that is, costs or benefits are

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[82] *Safety Light Corp.*, 41 N.R.C. at 457-458 (1995).

[83] *Sequoyah Fuels Corp. and General Atomics*, CLI-97-13, 46N.R.C. 195 (1997).

[84] Vermont Yankee Memorandum of Law Regarding Ratepayer Risk of Liability for Vermont Yankee Decommissioning Costs, Vermont Public Service Board Docket No. 6545, dated February 25, 2002, at pages 17 and 18.

> imposed on others yet are not paid for by those who impose them or receive them. More precisely, an externality is an effect of one economic agent's behavior on another's well-being, where that effect is not reflected in dollar or market transactions.[85]

Where there are such externalities, private decision-making will be inefficient. A company will tend to undervalue (or value at zero) the costs associated with its action that are borne by others. In the case of a nuclear power plant, the protection from liability may, for example, cause the operator to make decisions that undervalue the potential for long-term radioactive waste storage costs. Or, faced with operating decisions that involve tradeoffs between cost and safety, the owner may undervalue safety and make choices that strike the wrong balance. In these situations, because some of the risks are "external," the market outcome may be an inappropriate decision from a societal perspective — or an inefficient allocation of resources. Government policy efforts should aim to internalize externalities, in order to promote appropriate private decision-making and efficient resource allocation.

[85] Samuelson, Paul A. and William D. Nordhaus. 1989. Economics, 13th Edition. McGraw-Hill, at page 770.

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ATTACHMENT NO. 3

Entergy Corporation

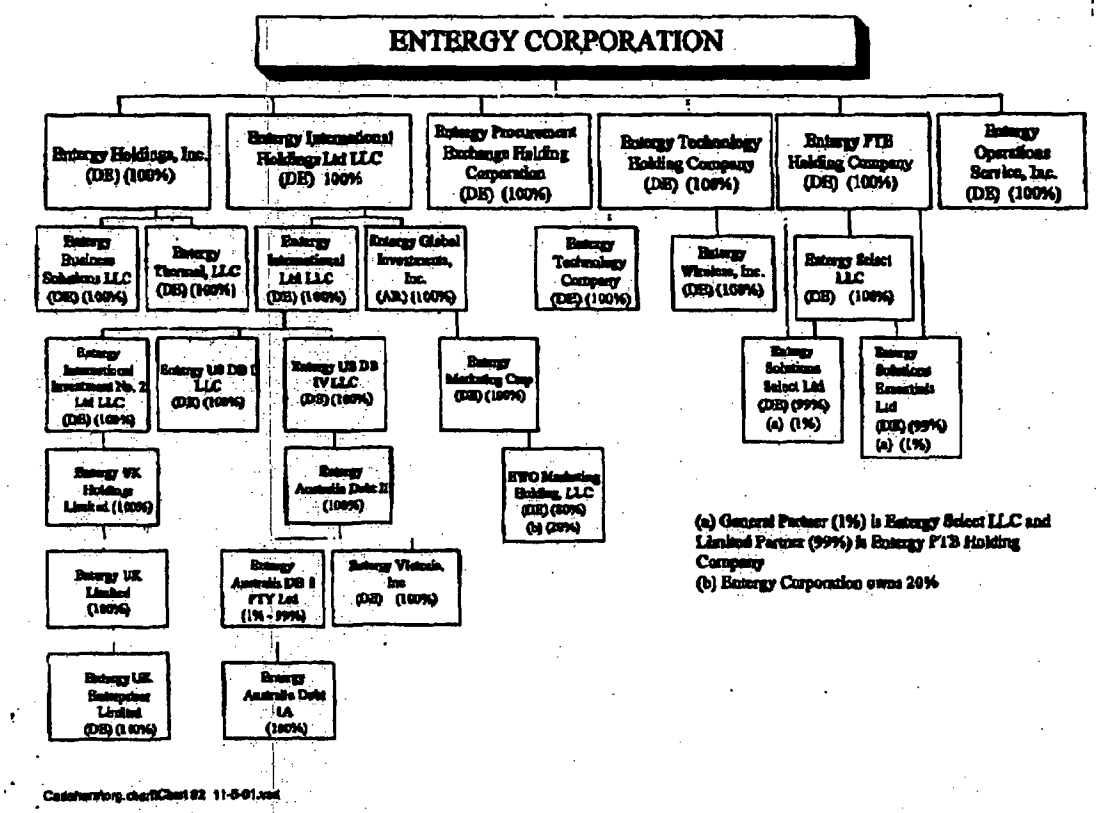
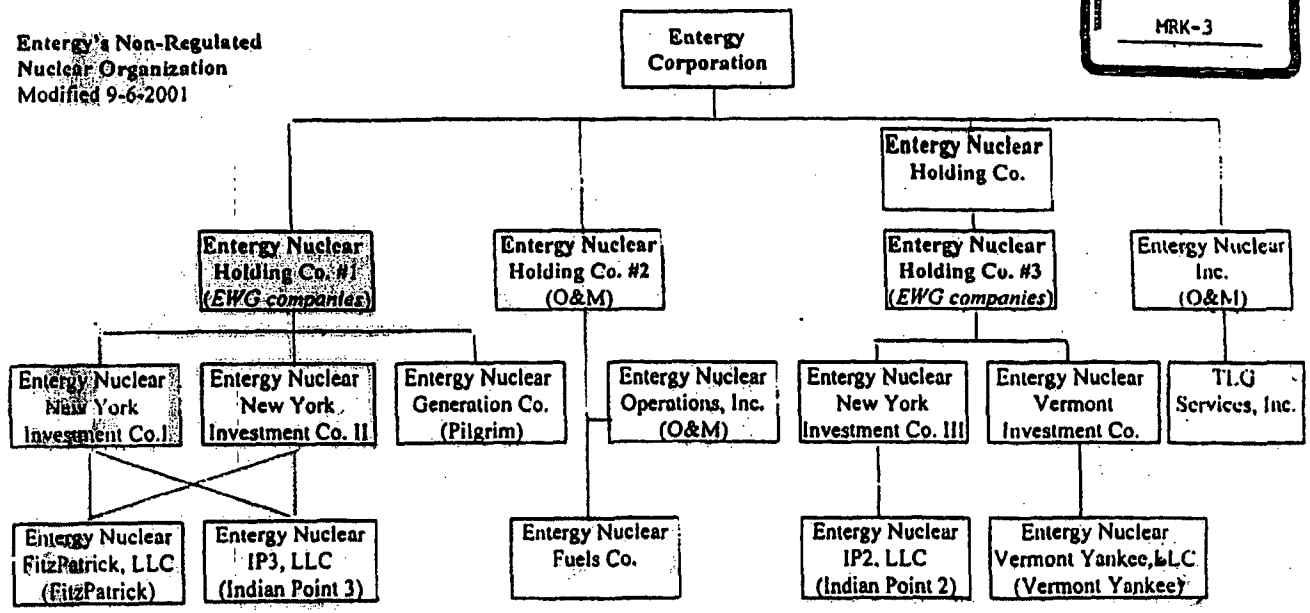


EXHIBIT
MRK-3

ATTACHMENT NO. 2

Entergy Corporation – Non-regulated Nuclear Organization



Exempt Wholesale Generator (EWG)
Operations and Maintenance (O&M)

EXHIBIT W

Entergy Holds New Orleans for Ransom

by Rita J. King, Special to Corp Watch
May 10th, 2006

cartoon by Khalil Bendib

Some New Orleanians desperately want, and fear, their utility bills.

But Tom Morgan, a DJ at New Orleans' local radio station WWOZ, hasn't seen one in months. While many residents have been hobbled by the cost of having a certified electrician reestablish electric and gas connections to their homes, some — like Morgan — have been unable to find out how much they owe and fear a gigantic bill they won't be able to pay, meaning – to add insult to injury – they face having their power cut off.

"I haven't gotten a bill since October," Morgan told CorpWatch. "I've called Entergy, but it still hasn't come. You shouldn't have to chase after people to pay your bills. A lot of people are confused and afraid."

Cutting and Running?

Entergy Corp. was the last Fortune 500 Company still based in New Orleans before Hurricane Katrina struck. Unless the federal government grants its wholly owned subsidiary, Entergy New Orleans, the $718 million it seeks to maintain and rebuild its gas and electricity infrastructure damaged in the storm, the utility might not be able to continue doing business in the city it currently supplies with gas and electric. It isn't that Entergy can't afford to rebuild; it's that it would rather keep its profits and let the federal government pick up the tab.

Entergy Corp. racked up $10 billion in revenues last year and has $29 billion in collective assets. On paper, there is no question Entergy could comfortably cover its losses and rebuild the infrastructure of its utility business in New Orleans. On May 2, Entergy announced that its first-quarter profit rose nearly 13 percent, as higher energy prices offset disrupted sales following last year's hurricanes. Entergy CEO J. Wayne Leonard received a $1.1 million bonus at the end of 2005, according to SEC records, which coincidentally works out to one dollar per Entergy customer in the Gulf Coast left without power in the weeks following the hurricane.

But the company's executives feel that if anyone should pay the cost of its getting back into business, it should be ratepayers and taxpayers, and not its own shareholders. And indeed, the government may have little choice but to give in to what critics characterize as blackmail or extortion – or leave a major American metropolis powerless.

Gordon Howald, utilities analyst with New York-based Natexis Bleichroeder, said he hasn't warned clients to sell off their Entergy stock. When Entergy first announced the possibility of bankruptcy, he says he thought the claim was a bluff.

"Entergy is a pretty successful company making a lot of money and here you have all these people who have lost their homes," he said. "Months later, they're still bickering. As time goes on, it becomes more difficult to get recovery, and ratepayers will have to pick up the rest."

Using the Federal Government as Insurance

The story of Entergy in New Orleans is a cautionary tale, critics say, of privatizing utilities as critical as gas and electricity in major population centers. Although Entergy is regulated by the New Orleans City Council, and its customer base is the local citizenry, as a publicly held company it ultimately answers first to its shareholders who want to maximize their profits. To that end Entergy has made broad use of limited liability laws to structure the company and its subsidiaries in a way that insulates shareholders from liabilities such as storms. The result is a system whereby the company's own customers and taxpayers nationwide foot the bill when something goes wrong.

Entergy has estimated its losses post-Katrina at just over $1 billion, including lost customers (many residents who fled may not come back), and miles of gas pipeline corroded by the saltwater that poured over the levees. The insurance the company carried covered $250 million in damages. The rest will likely come from inflated rates for those customers who remain in New Orleans, and from federal funds.

The Federal Energy Regulatory Commission (FERC), which has minimal oversight of Entergy, does not require companies like Entergy to carry insurance to cover losses from catastrophic events such as a hurricane, even though conventional wisdom has long considered a Katrina-sized storm and flood inevitable. The cost of the insurance it does carry may be passed on to ratepayers, according to FERC spokesman Bryan Lee.

Citing a "precedent" set by bailouts of ConEdison and the airline industry after 9/11, Entergy New Orleans is seeking a Community Development Block Grant (CDBG) to make up the difference. Without CDBG aid, Entergy New Orleans estimates that the average ratepayer's share of the losses comes to $8,943, which would be exacted in the form of a rate increase of at least 140 percent.

No Cost of Doing Business in Hurricane Alley

In the days after the storm, all public comments focused solely on restoration, as if the effort to rebuild New Orleans could be accomplished with nothing more than elbow grease and well-wishing. President George W. Bush cut an iconic pose in New Orleans' Jackson Square on September 15, 2005, with his sleeves rolled up for a prime-time television appearance as he stood in front of a statue of Andrew Jackson and vowed "one of the largest reconstruction efforts the world has ever seen."

Meltdown? Not Our Problem

An Entergy subsidiary, Entergy Nuclear Northeast, operates the Indian Point nuclear power plant in Westchester County, New York. In late March, a huge crowd assembled to meet with Nuclear Regulatory Commission (NRC) and Entergy representatives, who repeatedly claimed that the numerous radioactive leaks recently detected seeping from the plant and toward the Hudson River were nothing to worry about. But the crowd was concerned, largely because they knew Entergy was positioned to protect its assets by filing for bankruptcy in the event of a nuclear disaster. Where, they wondered, was the motivation to maintain safety standards, if not the specter of lost profits?

"Corporations like Entergy have the luxury of walking away from these disasters unscathed, while citizens who lose their homes and jobs will be saddled with the additional burden of having to pay for Entergy's mess," said Lisa Rainwater, Indian Point campaign director for watchdog environmental group Riverkeeper. "The new consumer bankruptcy law is yet another disturbing example of how Washington is more interested in protecting the financial interests of corporate America than the financial security of hard-working Americans. It's appalling."

What has been happening in New York with Entergy Nuclear may provide clues about what's ahead for New Orleans. Activists have been sounding the alarm there for years about the potential for a private company to cut and run with its profits, leaving devastation – and a massive bill – in its wake.

Indeed, the nuclear industry is no stranger to federal assistance. In July 2003, the Bush-Cheney Energy Bill was passed by Congress, and Republican Senator John McCain referred to it as the "no lobbyist left behind" bill. It was considered by many to be a bonanza for the

"Throughout the area hit by the hurricane," Bush pledged, "we will do what it takes, we will stay as long as it takes, to help citizens rebuild their communities and their lives." Two weeks later, on October 4 in the Rose Garden at the White House, Bush had a new philosophy: cut non-security spending to fund the recovery effort — when necessary.

"As the federal government meets its responsibilities, the people of the Gulf Coast must also recognize its limitations," Bush said. "The engine that drives growth and job creation in America is the private sector, and the private sector will be the engine that drives the recovery of the Gulf Coast."

Entergy New Orleans heard the siren call of a bailout.

In November of 2005, according to a Reuters

nuclear industry, including nearly $6 billion in operating tax credits, over a billion in decommissioning subsidies for aging plants and a 20 year extension of liability caps for accidents at nuclear plants. $3 billion in research subsidies and a nearly equivalent sum for construction subsidies was included for new plants, an effort that put Entergy out in the front of the pack. Entergy Corp. made bipartisan campaign contributions totaling $1,205,983 to federal candidates during the 2004 election cycle.

Gary Taylor, CEO of Entergy Nuclear, Inc. said on May 25, 2005 at a meeting of the Nuclear Energy Institute in Washington, DC, that "Entergy is enjoying a "full partnership" with the Department of Energy in its quest to pave the way for the next generation of nuclear reactors.

"We're putting in money and they're matching it on a dollar for dollar basis," Taylor is quoted in the July/August 2005 issue of Nuclear Plant Journal. "Clearly, they have a responsibility to ensure that this is going in the right direction." He also said that the "President is trying to say that we [the government] will provide some financial certainty if things fall apart."

On August 7, 2002, Synapse Energy Economics published a report, "Financial Insecurity: The Increasing Use of Limited Liability Companies and Multi-Tiered Holding Companies to Own Nuclear Power Plants," which was commissioned in part by Riverkeeper.

Former Commissioner of the NRC, Peter Bradford, wrote in his the foreword that the Synapse report "dissects a troublesome set of developments on the cusp between economic and safety regulation, namely, the arrangement of nuclear plant ownership into the limited liability subsidiaries of a few large companies. Because this arrangement has occurred during a period of lax and dispirited regulation, some important issues have not

report, Entergy unveiled a $3 billion plan that it said would "ensure liquidity" while it "awaits recovery." During a conference call with analysts at that time, Leonard said the company intended to be "relentless in recovery of storm costs."

"New Orleans is Entergy's home and we are absolutely dedicated to the city's reconstruction and resurrection," said Leonard immediately after the storm. "We are hopeful that we will be able to return home soon. Our ability to do that depends, of course, on a number of factors over which we do not have complete control."

"We are heartsick at the losses our communities and employees have suffered," said Curt L. Hebert, executive vice president, external affairs at Entergy Corp. in a public statement. "Even as we launch the largest power restoration in our country's history, we are equally concerned about reaching out to help our co-workers, families and neighbors restore their lives. Together we can and will rebuild and put this storm behind us."

Hebert is a product of the corporate-political revolving door. He was appointed chairman of FERC in 2001 by Bush, but stepped down months later, in September 2001, to take his position at Entergy Corp. He was in charge of lobbying the federal government for aid money after Katrina. That put him at odds with the chairman of the Gulf Coast Recovery and Rebuilding Council, Allan B. Hubbard, whose job it was to explain to Entergy that the feds would not be underwriting the company's New Orleans reconstruction effort.

In a letter dated November 16, 2005, Hebert noted he was "gravely disappointed," that the people of New Orleans would "suffer significantly" as a result of the Bush Administration's "fundamentally flawed" perspective. This stance, he charged, "repudiates the promise" made by Bush in Jackson Square.

been pursued effectively."

Years of "reckless undermining," he said, have now been "exposed in a series of financial collapses," among them a "ruinous mix of money, pressure, complexity and ideology."

"Regulating in this way is like driving drunk," Bradford went on. "Taxpayers, utility customers and power plant neighbors who thought themselves protected by firm requirements may one day wear the stunned expressions of Enron retirement plan holders or WorldCom investors."

According to Synapse, limited liability structure is an "effective mechanism" for transferring profits up the chain while creating a shield for the parent in case an unanticipated cost occurs at one of the plants.

"I also want to state clearly," Hebert wrote, "that [Entergy's shareholders and bondholders] have invested in a regulatory agency with the knowledge that the government regulates and protects their opportunity to earn a return."

With those words, Hebert inadvertently laid out the billion-dollar question: where do the boundaries of corporate welfare start and end?

In a response dated November 18, Hubbard pointed out Entergy's healthy bottom line while musing about the "inappropriate" nature of asking the federal government a handout while clasping a fistful of dollars.

"You told us that the ... board of the parent has a fiduciary duty not to take funds from other subsidiaries and use them to subsidize New Orleans," Hubbard wrote. "We respect the right of your board to decide how to allocate financial resources, such as last year's $909 million in earnings among various parts of Entergy Corp. We in turn believe it is inappropriate to transfer taxpayer resources to those investors after the fact for a risk they chose to take."

Prudent investors, he added, consider the risks inherent in any investment they make, including the risks of a natural disaster.

Ten days later, in a seven-page letter that smacked of one-upmanship, Hebert threatened that without "immediate federal assistance, it is unlikely that Entergy New Orleans can continue as a viable commercial entity." The threat was on the table: pay up or we pull up stakes.

Hebert also took exception to Hubbard's analysis, arguing that investment risk in a private company might be one thing, but risk in a publicly regulated utility quite another. He insisted that such public-private companies are by nature "entitled the opportunity to recover ... storm restoration costs."

Hebert says the obvious parallel and precedent was September 11. In the aftermath of that disaster, Congress passed the 2002 Supplemental Appropriation Act for Further Recovery From and Response to Terrorist Attacks on the United States. The $783 million in resulting CDBG funding included restoration of utility infrastructure for ConEdison, which, like Entergy New Orleans, is a publicly regulated utility and a subsidiary of a vast holding company. The airline industry, also, was offered a $5 billion bailout after 9/11 and a guarantee of up to $10 billion.

What Can Be Predicted Can Be Insured Against

But is a terrorist attack really a precedent for a natural disaster?

Not at all, said Howard Kunreuther, Cecilia Yen Koo Professor of Decision Sciences and Public

Policy and Co-Director of the Wharton Risk Management and Decision Processes Center at the University of Pennsylvania's Wharton School. Kunreuther has studied and written about the issue of insurance extensively, including an August 2005 report on "Terrorism Risk Financing in the United States," which outlines the many ways in which terrorism is a unique threat, and a January 2006 University of Pennsylvania Press publication, Has the Time Come for Comprehensive Natural Disaster Insurance? This work includes a chapter, "On Risk and Disaster: Lessons Learned from Hurricane Katrina ."

"9/11 is not a fair precedent for a natural disaster at all," Kunreuther told CorpWatch. "They're both very different in how they've been treated by the insurance industry. With a terrorist attack, you have less control. With a natural disaster, you can protect yourself to some degree."

Those who claim that Katrina was a completely unpredictable event may not have seen the many articles in publications ranging from National Geographic to a five-part series published pre-Katrina in the New Orleans Times-Picayune detailing the potential ramifications of just such a storm. And then there was Eric Berger's article in the Houston Chronicle on December 1, 2001, months after the Bush Administration's announcement of intent to downsize FEMA, in which it was reported that FEMA had declared the top three likeliest devastating emergencies in the United States to be a terrorist attack in New York City, a hurricane hitting New Orleans and a massive earthquake on the San Andreas fault.

Deregulation Comes Back to Roost

According to a May 2004 report from the United States General Accounting Office (GAO), limited liability companies such as Entergy Corp resulted from the deregulation of the electricity industry in the 1990s. "Like a partnership," the report said, "the profits are passed through and taxable to the owners ... like a corporation, it is a separate and distinct legal entity and the owners are insulated from personal liability for its debts and liabilities."

Such structures are made of loopholes the way some castles are made of sand—both, it turns out, can crumble under the sheer force of water.

"Entergy is a great example of how a company can shift liabilities to maximize profits while limiting liability," said Phillip Musegaas, senior policy analyst for the environmental watchdog group Riverkeeper, which is fighting to shut down the Entergy-owned and operated Indian Point nuclear power plants in Buchanan, New York. "Corporate restructuring is very sophisticated. They know their way out of regulation. They are way ahead of us."

All Profit, No Risk

It isn't "fair," said Entergy spokesman Stewart, to pass the cost of reconstructing Entergy New Orleans, the smallest subsidiary under the Entergy Corp umbrella, along to shareholders when the future fate of the company is still uncertain. Stewart explained that each subsidiary is a "separate business," and that each company is "protected from the burden" of picking up unexpected costs from the others.

"It would be irresponsible," Stewart told us. "It would not be prudent to invest shareholder money into the utility if there's no chance of recouping the money."

The corporation also has a "moral responsibility," he added, not to "risk the retirement funds of

employees when we don't know if customers are returning."

Utility investors rely on a "business model," he said, that allows for the utility to pass the costs of reconstruction and damage from natural events along to ratepayers. With the exception of totally decimated areas of New Orleans, power has been returned to "anyone who can take it." He likened the devastation in the region to that suffered by orange juice producers when frost nips the growing season short.

"The price of juice goes up," he said, "and that's the models investors have invested in here."

Elizabeth Raley, spokesperson for Entergy New Orleans, acknowledged that customers are "very upset."

"We're faced with many customers who have extremely high bills because the natural gas wells in the Gulf of Mexico were damaged," she said. "The market rate for gas is passed on to our customers. We're working with customers to help them find a way to pay their bills. It's challenging to serve our customers, but that's what we're doing."

New Orleanians are wary and suspicious of what Entergy may do next. "Our customers have been through life-altering situations," Raley said. "I wouldn't be surprised if some of them feel hostility simply because they are having a rough time."

It turns out that the hostility dates from long before the storm, evidenced by a series of local and federal lawsuits. In an SEC 10-K filing, Entergy Corp describes its relationship with the state of Louisiana as "particularly litigious." 480 class action lawsuits including 10,000 claims have been filed against Entergy's various Gulf Coast subsidiaries, including Entergy New Orleans, for damages allegedly caused by disposal of hazardous waste and "asbestos-related disease" by contractor employees who worked between the years 1950 and 1980 and claim to have been exposed to hazardous materials during that time. In March 2004, endangered brown pelicans were found dead near Entergy New Orleans' Michoud power plant intake structure and return trough. And then there were the ratepayer lawsuits brought on by the City Council, during which it was alleged in testimony filed over a period of years that customers had been overcharged by upwards of $100 million. When the matter was settled in April 2003, ratepayers were reimbursed $11.3 million when it was found that Entergy had been incorrectly calculating the cost of fuel and passing the error along to customers.

This had all been sorted out since, according to Clinton A. Vince, managing partner of Sullivan & Worcester's Washington DC office, which represents the New Orleans City Council, the regulatory agency responsible for oversight of Entergy New Orleans. Vince supports the idea of a CDBG bailout.

"I won't second-guess them on their system. Nobody ever could have anticipated a flood of this magnitude," he said when asked why pipes that were susceptible to corrosion salt water were buried underground in a city that lies below sea-level. "This was not a reasonably foreseeable event."

"Entergy needs a reorganization plan," Vince said. "The content of that plan will change drastically based on CDBG funds. There would be dire consequences if Entergy walked away. We need the utility to stay there and rebuild the system. It's a unique situation—the worst disaster for a utility in the history of the country. People have lost everything, and it would be unrealistic

and unfair to pass those costs along [to ratepayers]. People would have no incentive to return."

* * *

On March 30, 2006, Donald Powell, the Bush administration's Gulf Coast recovery coordinator, said the revival of the Crescent City could take up to a quarter of a century and also hinges on factors that are "out of our control." The amount of total funds that will be allotted by the state and federal governments, Powell said, is still up in the air.

Will Entergy choose – or be forced – to cut and run from New Orleans?

"Our plan is to stick it out," Entergy New Orleans' spokeswoman Raley told Corpwatch. "We're very hopeful that our funds will come through, and we'll be able to continue to operate."