UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC
Entergy Nuclear Indian Point 3, LLC and
Entergy Nuclear Operations, Inc.

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Docket Nos. 50-247-LR and 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

NEW YORK STATE NOTICE OF INTENTION TO PARTICIPATE AND PETITION TO INTERVENE AND SUPPORTING DECLARATIONS AND EXHIBITS

Volume I of II

Filed on November 30, 2007

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DECLARATION OF DAVID LOCHBAUM

David Lochbaum, hereby declares under penalty of perjury that the following is true and correct:

1. Attached hereto and made a part of this sworn statement are a report prepared by me regarding the importance of compliance with 10 C.F.R. § 50.71(e) and lack of compliance by Indian Point Energy Center with that requirement, a chart prepared by me based on the information contained in Nuclear Regulatory Commission records and the UFSAR filed by IP 2 and IP 3 with its LRA in this case and my current CV. The report, chart and CV are true and correct to the best of my personal knowledge.

Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated: November 27, 2007
Knoxville, Tennessee

David Lochbaum

INDIAN POINT ENERGY CENTER, 10 CFR 50.71(E), AND LICENSE RENEWAL

INTRODUCTION OF 10 CFR 50.71(e)

In May 1980, the NRC issued a final rule, 10 CFR 50.71(e), applicable to all licensees of operating power reactors like Indian Point Units 2 and 3. That final rule stated:

> *Each person licensed to operate a nuclear power reactor pursuant to the provisions of § 50.21 or § 50.22 shall update periodically, as provided in paragraphs (e)(3) and (e)(4) of this section, the final safety analysis report (FSAR) originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed.*
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> and
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> *The updated FSAR shall be revised to include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question; and all analyses of new safety issues performed by or on behalf of the licensee at Commission request. The updated information shall be appropriately located within the FSAR.*[1]

This rule had the effect of requiring the FSAR[2] to become a "living document" that is periodically updated to incorporate information regarding applicable modifications to the facility and procedures.

In December 1980, the NRC used its generic communications process to remind its licensees about their obligations under the recently promulgated rule. By Generic Letter 80-110, the NRC notified its licensees that:

> *The Commission approved the rule 50.71(e) (copy enclosed) entitled "Periodic Updating of Final Safety Analysis Reports" and published the rule in the Federal Register on May 9, 1980. The rule became effective on July 22, 1980.*
>
> and
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> *For non-SEP* [Systematic Evaluation Program] *plants, the rule requires submittal of the updated FSAR within 24 months of either July 22, 1980, or the date of issuance of the operating license, whichever is later.*

The NRC's Systematic Evaluation Program (SEP) was an initiative begun in 1977 seeking to compare the licensing bases for 11 older nuclear power reactors (Dresden Units 1&2, Yankee Rowe, Big Rock Point, San Onofre Unit 1, Connecticut Yankee, LaCrosse, Oyster Creek, Ginna, Millstone Unit 1, and Palisades) to current safety regulations.[3] Because Indian Point Units 2 and

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[1] U.S. Nuclear Regulatory Commission, Final Rule, "Periodic Updating of Final Safety Analysis Reports," *Federal Register*, Vol. 45, No. 92, May 9, 1980, pp. 30615-30616.

[2] Over the years, the NRC and industry have inconsistently applied various terms to the FSAR. Some have used FSAR to refer to the original document submitted, and amended, for the initial operating license application and used Updated Final Safety Analysis Report (UFSAR) or Updated Safety Analysis Report (USAR) to refer to the periodically updated FSAR. For the purposes of this paper, all three terms are considered interchangeable and FSAR will be used outside of quoted material.

[3] U.S. Nuclear Regulatory Commission, Press Release No. 77-196, "NRC Staff to Begin New Systematic Evaluation of 11 Operating Nuclear Power Facilities," November 17, 1977.

3 were non-SEP reactors, 10 CFR 50.51(e) required the first update to the FSAR to be submitted to the NRC on or before July 22, 1982.

RE-EMPHASIS OF 10 CFR 50.71(e)

Nearly 15 years later, the NRC and the nuclear industry became aware of compliance problems with the 10 CFR 50.71(e) requirements. While the most prominent compliance problems surfaced at the Millstone nuclear plant in 1996, the problems extended beyond this one facility and prompted the NRC and the nuclear industry to take steps to remedy the situation.

The nuclear industry, via the Nuclear Energy Institute (NEI), developed guidance document NEI 98-03 Rev. 1 in June 1999 outlining the steps needed to comply with 10 CFR 50.71(e):

> *Inspections in 1996-1997 by the NRC and licensees identified numerous discrepancies between UFSAR information and the actual plant design and operation. These findings have raised questions about possible noncompliance with 10 CFR 50.71(e). The industry has developed this guidance in recognition of the importance of the UFSAR, the need to comply with 10 CFR 50.71(e) update requirements, and the need for UFSARs to be consistent with the plant design and operation.*[4]

The NRC issued Regulatory Guide 1.181 in September 1999 to clarify its regulatory position on updating the FSARs:

> *As a result of lessons learned from the Millstone experience and other initiatives related to UFSARs, the NRC has determined that additional guidance regarding compliance with 10 CFR 50.71(e) is necessary. ... In a staff requirements memorandum dated May 20, 1997, the Commission directed the staff, in part, to issue guidance for complying with 10 CFR 50.71(e) so that UFSARs are updated to reflect changes to the design bases and to reflect the effects of other analyses performed since original licensing that should have been included under 10 CFR 50.71(e). This regulatory guide provides the guidance requested by the May 20, 1997, staff requirements memorandum.*
>
> and
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> *The objectives of 10 CFR 50.71(e) are to ensure that licensees maintain the information in the UFSAR to reflect the current status of the facility and address new issues as they arise, so that the UFSAR can be used as a reference document in safety analyses.*[5]

Among other things, the NRC formally endorsed NEI 98-03 in Regulatory Guide 1.181 as an acceptable method of complying with 10 CFR 50.71(e), although they readily acknowledged that licensees may employ other methods to comply:

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[4] Nuclear Energy Institute, NEI 98-03 Rev. 1, "Guidelines for Updating Final Safety Analysis Reports," June 1999, page 1.

[5] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," September 1999, pp. 1-2. Available online at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-181/01-181.pdf and from the NRC's Agencywide Document Access and Management System (ADAMS) via accession number ML003740112).

Revision 1 of NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," dated June 1999, provides methods that are acceptable to the NRC for complying with the provisions of 10 CFR 50.71(e). ... Licensees may use methods other than those proposed in Revision 1 of NEI 98-03 to meet the requirements of 10 CFR 50.71(e). The NRC will determine the acceptability of other methods on a case-by-case basis. [6]

NEI 98-03 Rev. 1 described the FSAR's role:

UFSARs provide a description of each plant and, per the Supplementary Information for the FSAR update rule, serve as a "reference document to be used for recurring safety analyses performed by licensees, the Commission, and other interested parties." The UFSAR is used by the NRC in its regulatory oversight of a nuclear power plant, including its use as a reference for evaluating license amendment requests and in the preparation for and conduct of inspection activities. For licensees, portions of the UFSAR are used as a reference in evaluating changes to the facility and procedures under the 10 CFR 50.59 change process. The UFSAR also serves to provide the general public a description of the plant and its operation.[7]

NEI 98-03 Rev. 1 described the updates to FSARs required by 10 CFR 50.71(e):

Based on analysis of 10 CFR 50.34(b), UFSAR updates should contain the following basic types of information concerning new requirements and information developed since the UFSAR was last updated that are required to be reflected in the UFSAR under 10 CFR 50.71(e):

- *new or modified design bases*
- *summary of new or modified safety analyses*
- *UFSAR description sufficient to permit understanding of new or modified design bases, safety analyses, and facility operation* [8]

NEI 98-03 Rev. 1 defined the "safety analyses" covered by the second bullet to be:

Safety analyses are analyses performed pursuant to Commission requirement to demonstrate the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1) or 10 CFR 100.11. Safety analyses are required to be presented in the UFSAR per 10 CFR 50.34(b) or 10 CFR

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[6] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," September 1999, page 3. Available online at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-181/01-181.pdf and from the NRC's Agencywide Document Access and Management System (ADAMS) via accession number ML003740112).

[7] Nuclear Energy Institute, NEI 98-03 Rev. 1, "Guidelines for Updating Final Safety Analysis Reports," June 1999, page 3.

[8] NEI 98-03 Rev. 1, page 4.

50.71(e) and include, but are not limited to, the accident analyses typically presented in Chapter 14 or 15 of the UFSAR. [9]

NEI 98-03 Rev. 1 described what constituted "new or modified safety analyses" (vice restated safety analyses) and the related level of detail issue for summaries of new or modified safety analyses:

> *Licensees should evaluate the effects of analyses or similar evaluations performed by licensees in response to plant-specific NRC requests or NRC generic letters or bulletins. NRC-requested analyses and evaluations must be reflected in UFSAR updates only if, on the basis of the results of the requested analysis of evaluation, the licensee determines that the existing design bases, safety analyses or UFSAR description are either not accurate or not bounding or both. The existing design bases, safety analyses and UFSAR description must be updated to reflect the new information, as appropriate.* [10]
>
> and
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> *While not explicitly addressing the level of detail required for FSARs, 10 CFR 50.34(b)(2) required that the original FSARs include:*
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> > *... description and analysis of the structures, systems, and components of the facility, with emphasis on performance requirements, the bases, with technical justification therefore, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.* [11]

NEI 98-03 Rev. 1 contained case studies to assist licensees decide what did and did not constitute new or modified safety analyses to be summarized in the FSARs. One case study covered the evaluation required by the NRC by Bulletin 88-04:

> *A change to the safety injection system was initiated to address an operability concern identified in NRC Bulletin 88-04, "Potential for Safety-Related Pump Loss." An evaluation of safety injection pump minimum-flow lines resulted in an increase in the recommended minimum-flow rate to preclude hydraulic instability at low flow conditions and assure pump operability. ... Because sufficient minimum-flow is necessary to ensure the system is able to perform its intended safety function, the UFSAR description associated with the safety injection system should be modified to include a discussion of the minimum-flow function as it relates to maintaining operability of the safety injection pumps. In some cases, this may entail adding UFSAR discussion of the minimum-flow function where none previously existed.* [12]

NRC RE-VISITS 10 CFR 50.71(E) FOR INDIAN POINT UNITS 2 AND 3

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[9] NEI 98-03 Rev. 1, page 2.
[10] NEI 98-03 Rev. 1, page 7.
[11] NEI 98-03 Rev. 1, page 8.
[12] NEI 98-03 Rev. 1, page 9.

On October 9, 1996, the NRC requested pursuant to 10 CFR 50.54(f) that the licensees of Indian Point Units 2 and 3 submit material to the NRC, under oath or affirmation, regarding the adequacy and availability of design bases information.[13] The NRC expressly informed the Indian Point licensees that "the NRC staff has found that some licensees have failed to ... assure that UFSARs properly reflect the facilities." [14] The NRC described the nexus between these failures and public safety:

> *Of particular concern is whether licensee programs are consistent with and are being maintained in accordance with their design bases. The extent of the licensees' failures to maintain control and to identify and correct the failures in a timely manner is of concern because of the potential impact on public health and safety should safety systems not respond to challenges from off-normal and accident conditions.* [15]

The NRC requested the licensees to take five actions, the first being to provide the NRC with a:

> *Description of engineering design and configuration control processes, including those that implement 10 CFR 50.59,* ***10 CFR 50.71(e)****, and Appendix B to 10 CFR Part 50* [16] [emphasis added]

By letter dated February 13, 1997, the licensee for Indian Point Unit 2 responded to the NRC's 10 CFR 50.54(f) request. [17] The licensee described its process for updating the FSAR:

> *The 10 CFR 50.59 evaluations are used to identify updates to the Updated Final Safety Analysis Report (UFSAR). Updates to the UFSAR include the effects of changes made to the facility or procedures described in the USAR, Safety Evaluations performed in support of requested license amendments or conclusions that changes have not involved an unreviewed safety question (USQ) (10 CFR 50.59 process).* [18]

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[13] U.S. Nuclear Regulatory Commission letters from James M. Taylor, Executive Director for Operation, to E. R. McGrath, Consolidated Edison Company of New York, Inc., and Robert G. Schoenberger, President and Chief Executive Officer, Power Authority of the State of New York, "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Basis Information," October 9, 1996. Available from the NRC's Public Document Room via accession numbers 9610110273 and 9610110057.

[14] Ibid.

[15] Ibid, page 5.

[16] Ibid, page 6.

[17] Consolidated Edison Company of New York, Inc. letter from Stephen E. Quinn, Vice President, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information, NRC Letter from James M. Taylor to Eugene McGraht dated October 9, 1996," February 13, 1997. Available from the NRC's Public Document Room via accession number 9702190330.

[18] Consolidated Edison Company of New York, Inc. letter from Stephen E. Quinn, Vice President, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information, NRC Letter from James M. Taylor to Eugene McGraht dated October 9, 1996," February 13, 1997, page 3.1-3. Available from the NRC's Public Document Room via accession number 9702190330.

By letter dated February 7, 1997, the licensee for Indian Point Unit 3 responded to the NRC's 10 CFR 50.54(f) request. [19] The licensee described its process for updating the FSAR:

> *The process for updating the FSAR is controlled by procedure NLP-3 "FSAR Updates." This procedure requires that the FSAR is updated to reflect plant modifications, changes to procedures described in the FSAR, 10CFR50.59 Safety Evaluations, Technical Specification Amendments, NRC correspondence, and to reflect the on resolution of discrepancies* [sic]. [20]

The FSAR updating processes for Indian Point Units 2 and 3 were administered by different licensees at that time, but the processes were very similar. Both relied on the 10 CFR 50.59 process to trigger updates to the FSARs. This regulation controls when and under what conditions licensees "*may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated)*" without first obtaining NRC approval. The 10 CFR 50.59 processes do not trigger updates to the FSARs for safety analyses performed at the NRC's request unless those safety analyses also involve a plant modification or procedure revision.

RELEVANCE OF 10 CFR 50.71(e) FOR LICENSE RENEWAL OF INDIAN POINT UNITS 2 AND 3

Federal regulation 10 CFR 50.71(e) requires NRC licensees of operating nuclear reactors like Indian Point Units 2 and 3 to periodically update the FSARs for their facilities to include applicable information from safety analyses performed at the NRC's request.

NEI's guidance document 98-03 Revision 1 describes a methodology for updating FSARs to conform to 10 CFR 50.71(e) requirements. This guidance defines "safety analyses" and details what constitutes applicable information from safety analyses performed at the NRC's request to be incorporated into the FSARs.

NRC's Regulatory Guide 1.181 endorsed NEI 98-03 Rev. 1 as an acceptable means for conforming to the 10 CFR 50.71(e) requirements, but provided licensees the option of establishing an alternative means of conformance for the NRC to review and accept.

NRC's generic correspondence program uses Regulatory Issue Summaries, Information Notices, Generic Letters, and Bulletins to make licensees aware of relevant operating experience and to require licensees to take certain actions based on that operating experience. Regulatory Issue Summaries and Information Notices involve administrative (e.g., scheduling testing of

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[19] New York Power Authority letter from Harry P. Salmon, Jr., Chief Nuclear Officer – Acting, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information," February 7, 1997. Available from the NRC's Public Document Room via accession number 9702120120.

[20] New York Power Authority letter from Harry P. Salmon, Jr., Chief Nuclear Officer – Acting, to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information," February 7, 1997, page 22 of attachment 1. Available from the NRC's Public Document Room via accession number 9702120120.

candidates for operator licenses) and information awareness matters. Generic Letters and Bulletins typically involve actions NRC requests licensees to take.

I reviewed NRC Generic Letters and Bulletins issued since 1982, corresponding to when 10 CFR 50.71(e) required the Indian Point 2 and 3 licensees to begin submitting periodic updates to the FSARs to include applicable information from safety analyses requested by the NRC via these generic correspondence documents. The generic correspondence documents included in this review are listed in Table 1.

For Indian Point Units 1, 2, and 3, I reviewed the licensee responses to the NRC generic correspondence. I also reviewed the latest FSARs for references to this NRC generic correspondence. The responses I reviewed are also listed in Table 1 along with references found within the FSARs.

I also reviewed the license renewal application for Indian Point Units 2 and 3 for references to the NRC generic correspondence. Table 1 also indicates the NRC generic correspondence references found in the license renewal application.

I have summarized below some of the Table 1 entries. Generally, these summaries indicate what safety analyses were performed by the Indian Point licensees in response to NRC generic correspondence and how that information was, or was not, reflected within the FSARs.

NRC Bulletin 82-02

The NRC issued Bulletin 82-02 on June 2, 1982, to licensees of operating pressurized water reactors regarding an age-related degradation mechanism (corrosion) affecting bolts and studs in the reactor coolant pressure boundary. Specifically, this bulletin informed licensees of significant corrosion of the closure studs for the reactor coolant pumps at Fort Calhoun and of closure studs for a steam generator primary manway at Maine Yankee. This bulletin required licensees to take appropriate steps to reduce the likelihood of similar corrosion problems at their facilities.

The Indian Point Unit 2 licensee responded to the NRC on August 2, 1982, reporting that wastage in 8 of the 24 closure studs for reactor coolant pump 23 had occurred due to leakage from the pump's main flange. The Indian Point Unit 2 licensee's response also informed the NRC that the reactor coolant pump insulation had been changed from blanket insulation to a preformed metal type to preclude boric acid buildup, and resulting corrosion, in event of future leaks. NRC Bulletin 82-02 is not mentioned in the Indian Point Unit 2 FSAR. The design change to the reactor coolant pump insulation type made to prevent significant age-related degradation (corrosion) is not mentioned in the Indian Point Unit 2 FSAR.

The licensee's August 2, 1982, response to the NRC for Bulletin 82-02 credited a design change in the type of insulation applied to the reactor coolant pumps in preventing significant age-related degradation from boric acid accumulation. This modification to the plant and its bases was not incorporated into the Indian Point Unit 2 FSAR. The role of the preformed metal type insulation in minimizing corrosion, credited in the licensee's response to NRC Bulletin 82-02, is not mentioned in the Indian Point license renewal application.

NRC Bulletin 84-03

The NRC issued Bulletin 84-03 on August 24, 1984, to licensees of operating reactors regarding the failure of the refueling cavity water seal at Connecticut Yankee that resulted in approximately 200,000 gallons of water flooding containment. This bulletin required licensees to evaluate the potential for and consequences of a refueling cavity water seal failure at their facilities.

On March 31, 1987, the licensee for Indian Point Unit 2 submitted to the NRC safety analyses performed in response to Bulletin 84-03. These safety analyses included evaluations of postulated breaks in 4-inch and 6-inch diameter piping (assumed to occur individually) that results in drainage of water from the refueling cavity. The purpose of these evaluations was to determine if sufficient time existed for the operators to move an irradiated fuel assembly from an elevated position into a safe location before the lowering water level uncovered it. An estimated maximum time of 1.1 hours for the operators to re-position an elevated irradiated fuel assembly was used in the evaluations, which concluded that the calculated draindown times provided at least that amount of time. The submittal also apprised the NRC of the steps the operators would take, in addition to relocating an elevated irradiated fuel assembly, in response to a refueling cavity draindown event. Section 9.5.3.2, Malfunction Analysis, of the Indian Point Unit 2 FSAR states "Various potential failures, which could create paths for drainage from the refueling cavity, have been considered. A plant procedure defines actions to deal with these postulated events." Section 14.2.1, Fuel-Handling Accidents, of the Indian Point Unit 2 FSAR stated "Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit." This section summarizes the results from evaluations of three postulated accident: (1) drop of a fuel assembly in the fuel handling building, (2) drop of a fuel assembly inside containment, and (3) drop of a spent fuel storage cask.

The Indian Point Unit 2 FSAR has not been updated to reflect the safety analyses performed in response to NRC Bulletin 84-03. The FSAR currently contains a discussion of a fuel handling accident in which fuel rods are damaged from the postulated dropping of a fuel assembly. But the FSAR does not contain a discussion of the other credible fuel handling accident scenario evaluated by the licensee in response to NRC Bulletin 84-03; namely, that fuel rods are damaged by the fuel assembly remaining in place while the refueling cavity water level drops. The associated operator response times and procedural steps to prevent fuel damage in event of water inventory loss have not been incorporated into the Indian Point Unit 2 FSAR.

By letter dated November 27, 1984, the licensee for Indian Point Unit 3 submitted to the NRC safety analyses performed in response to Bulletin 84-03. These safety analyses included evaluations of various refueling cavity water seal failure modes such as deflection of the seal due to hydrostatic pressure, compressive forces that might push the seal through the annular gap between the reactor vessel flange and the reactor cavity floor, and damage resulting from a fuel assembly dropping onto the seal and deflating it. The safety analyses additionally considered the response time for operators to mitigate a refueling cavity draindown. The submittal informed the NRC that operators can close the fuel transfer tube gate valve in approximately 20 minutes to isolate the fuel building from the reactor cavity, that operators can move the fuel transfer cart from the fuel building to the containment in approximately 5 minutes, and that operators can

lower a fuel assembly in the upender from the vertical position to the horizontal position in approximately 2 minutes. Section 9.5.3, System Evaluation, of the Indian Point Unit 3 FSAR states "An analysis is presented in Chapter 14 concerning damage to one complete outer row of fuel rods in an assembly. This accident is assumed as a conservative limit for evaluating environmental consequences of a fuel handling accident."

The Indian Point Unit 3 FSAR has not been updated to reflect the safety analyses performed in response to NRC Bulletin 84-03. The FSAR currently contains a discussion of a fuel handling accident in which fuel rods are damaged from the postulated dropping of a fuel assembly. But the FSAR does not contain a discussion of other credible fuel handling accident scenarios evaluated by the licensee in response to NRC Bulletin 84-03; namely, (a) that fuel rods are damaged by the fuel assembly remaining in place while the refueling cavity water level drops, and (b) that a dropped fuel assembly damages the refueling cavity water seal causing a loss of water inventory. The associated operator response times and procedural steps to prevent fuel damage in event of water inventory loss have not been incorporated into the Indian Point Unit 3 FSAR.

NRC Bulletin 87-01

The NRC issued Bulletin 87-01 on July 9, 1987, to licensees of operating nuclear reactors regarding a December 1986 event at the Surry nuclear plant that resulted in the deaths of four workers. Erosion/corrosion of a carbon steel pipe in the feedwater system caused it to rupture and release a two-phase mixture. This bulletin required licensees to take actions to prevent recurrence of this failure at their facilities.

By letter dated September 11, 1987, the licensee for Indian Point Unit 2 submitted its response to the bulletin to the NRC. The Indian Point Unit 2 licensee informed the NRC "*As a result of the Surry event, we have augmented our inspection program to include the following single phase systems: the main feedwater system, the condensate system, the heater drain pump discharge piping and the auxiliary feedwater system.*" The Indian Point Unit 2 licensee also informed the NRC "*We are expanding our high energy pipe inspection program. In addition to the extraction steam program, the following systems are being added to that program: condensate, feedwater, moisture separator drains, feedwater heater drains, steam generator blowdown.*" Section 10.4, Tests and Inspections, of the Indian Point Unit 2 FSAR states "In response to NRC IE Bulletin 87-01, an inspection program has been established for piping and fittings in the extraction steam, turbine crossunder, heater drain pump discharge, condensate, feedwater, and auxiliary feedwater systems. UT inspections are utilized to evaluate wall thickness at locations considered to be most susceptible to erosion/corrosion."

By letter dated September 15, 1987, the licensee for Indian Point Unit 2 submitted its response to the bulletin to the NRC. The Indian Point Unit 3 licensee informed the NRC "*As a direct result of the Surry event and other industry reported failures in single phase systems, the Authority undertook an expanded inspection program during the 1987 refueling outage.*" The discussion of erosion/corrosion piping degradation mechanisms and associated inspection regimes in the Indian Point Unit 3 FSAR is limited to steam generator tubes, service water system piping, and emergency core cooling system piping and fails to describe the inspection scope revisions made in response to NRC Bulletin 87-01.

NRC Generic Letter 87-12

The NRC issued Generic Letter 87-12 on July 9, 1987, to licensees of pressurized water reactors like Indian Point Units 2 and 3 regarding lessons learned from a loss of residual heat removal (RHR) cooling during midloop operation at Diablo Canyon. The bulletin required licensees to describe design features and procedures at their facilities that can prevent and/or mitigate loss of cooling events during midloop operations at their facilities.

By letter dated September 29, 1987, the licensee for Indian Point Unit 2 informed the NRC that "during RCS draindown, the Residual Heat Removal (RHR) System complies with the licensing basis for Indian Point Unit 3 as expressed in the FSAR and the Technical Specifications." The licensee went on to inform the NRC that "recognizing the potential significant of the Containment integrity issues addressed in the generic letter, we have conservatively analyzed offsite radiological consequences of RCS fluid boiloff without Containment integrity" and that the result from this safety analysis led the license to "prohibit draindown of the RCS to the water level where the potential for vortexing of RHR can occur unless the radioactivity level in the primary coolant is at an acceptable limit as defined in the attached analyses." No reference to Generic Letter 87-12 exists in the Indian Point Unit 2 FSAR and no discussion of these safety analyses and operational restrictions was found.

By letter dated September 21, 1987, the licensee for Indian Point Unit 3 provided the NRC with its response to Generic Letter 87-12. Section 4.3.7, Cold Shutdown RCS Level Indication, of the Indian Point Unit 3 FSAR describes the water level instrumentation installed in response to Generic Letters 87-12 and 88-17 – with explicit references to these documents – to monitor reactor coolant system conditions during cold shut down.

NRC Generic Letter 88-05

The NRC issued Generic Letter 88-05 on March 17, 1988, to licensees of operating pressurized water reactors including Indian Point Units 2 and 3 regarding events where reactor coolant leakage below the technical specification limits caused degradation of carbon steel components it contacted. The NRC reported:

> *In light of the above experience* [boric acid degradation at Turkey Point Unit 4, Salem Unit 2, and Fort Calhoun], *the NRC believes that boric acid leakage potentially affecting the integrity of the reactor coolant pressure boundary should be procedurally controlled to ensure continued compliance with the licensing basis. We therefore request that you provide assurances that a program has been implemented consisting of systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure or gross rupture.*
>
> and
>
> *The request that licensees provide assurances that a program has been implemented to address the corrosive effects of reactor coolant system leakage at less than technical specification limits constitutes a new staff position. Previous staff positions have not considered the corrosion of external surfaces of the reactor coolant pressure boundary. Based on the frequency and continuing pattern of significant degradation of the reactor*

coolant pressure boundary that was discussed above, the staff now concludes that in the absence of such a program compliance with General Design Criteria 14, 30, and 31 cannot be ensured.

By letter dated May 31, 1988, the licensee for Indian Point Unit 2 submitted a response to the NRC for Generic Letter 88-05. The licensee informed the NRC that its inspection program for boric acid corrosion developed in response to NRC Bulletin 82-02 "*has since been expanded to cover more than 350 mechanical connections.*" Section 4.2.7.3, Locating Leaks, of the Indian Point Unit 2 FSAR mentions that "the presence of boric acid crystals near the leak" makes visual observation a method of locating sources of escaping steam of water. Sections 6.7.1.2.1.3, Releases to the Containment Environment; 6.7.1.2.8, Steam Generator Blowdown Liquid Sample Monitor; 6.7.1.2.9, Residual Heat Removal Loop; 6.7.1.2.10, Recirculation Loop; and 6B.0, Operational Experience; contain similar discussions. No mention of Generic Letter 88-05, a boric acid corrosion control program, or an inspection program of mechanical components for borated water leakage and/or boric acid accumulation was found within the Indian Point Unit 2 FSAR.

By letter dated June 1, 1988, the licensee for Indian Point Unit 3 submitted a response to the NRC for Generic Letter 88-05. The licensee informed the NRC that it had revised procedures at Indian Point Unit 3 in response to the generic letter. The licensee informed the NRC that these steps included requiring "*prompt repair and clean-up* [of boric acid] *when the component can be readily made available for maintenance activities*" and "*an engineering evaluation for continuing operability in those instances where prompt corrective action is impractical.*" No mention of Generic Letter 88-05, a boric acid corrosion control program, or an inspection program of piping and components for borated water leakage and/or boric acid accumulation was found within the Indian Point Unit 3 FSAR.

In March 2002, workers at the Davis-Besse nuclear plant in Ohio discovered extensive degradation to the reactor vessel head caused by boric acid corrosion. The Davis-Besse licensee had also received Generic Letter 88-05 and committed to the NRC to implement a boric acid corrosion control program. A description of that boric acid corrosion control program was not incorporated into the Davis-Besse FSAR. Neither workers nor NRC inspectors detected over the intervening decade that the boric acid corrosion control program was not being implemented. Had the FSAR been updated to reflect the licensee's response to Generic Letter 88-05 as 10 CFR 50.51(e) requires, the likelihood would have been reduced that plant workers and NRC inspectors remained unaware of the implementation deficiency across this protracted period.

<u>NRC Generic Letter 88-17</u>

The NRC issued Generic Letter 88-17 on October 17, 1988, to licensees of operating pressurized water reactors including Indian Point Units 2 and 3 requiring actions to be taken to protect against fuel damage and release of radioactivity to the environment caused by reactor coolant system draindown during cold shut down. This generic letter required licensees to "*Implement procedures and administrative controls that reasonably assure that containment closure will be achieved prior to the time at which a core uncovery could result from a loss of DHR* [decay heat removal] *coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory.*"

By letters dated February 3, 1989, August 22, 1990, September 20, 1991, and July 28, 1997, the licensee for Indian Point Unit 2 responded to the NRC regarding Generic Letter 88-17. The licensee informed the NRC that it was installing two separate and diverse reactor coolant system (RCS) water level monitoring systems, adding a control room indicator for operators to monitor residual heat removal (RHR) flow conditions, revising procedures to de-energize two open motor-operated valves (RHR isolation valves MOV-730 and MOV-731) when RHR is operating, establishing a vent pathway prior to RCS draindown to the level where RHR pump vortexing is possible, and adding an alarm function for the RCS water level monitoring system. No mention of Generic Letter 88-17 was found within the Indian Point Unit 2 FSAR. Section 5.1.4.2.3, Equipment and Personnel Access Hatches, discusses containment integrity during refueling, but only when "*the Reactor Coolant System elevation >66 feet (i.e., **not** in reduced inventory)*" [emphasis in original] which is not the configuration of concern in Generic Letter 88-17. Section 9.3.2.2, Residual Heat Removal Loop, states that "*Instrumentation has been provided in the control room to monitor RHR and reactor coolant system level when the system is cooled and depressurized*" and describes the design features of this level monitoring instrumentation. Section 7.5.2.1.16, Reactor Coolant System Pressure, describes how pressure instruments provide redundant interlock signals to RHR isolation valves MOV-730 and MOV-731 to prevent them from opening at high RCS pressure, but there is no mention in the FSAR about de-energizing these valves in the open position when RHR is operating during cold shut down with the reactor coolant system partially drained.

NRC Bulletin 94-01

The NRC issued Bulletin 94-01 on April 14, 1994, to licensees with irradiated fuel stored in spent fuel pools at permanently shut down nuclear reactors including Indian Point Unit 1 regarding lessons learned from an event at Dresden Unit 1 where cold weather caused water to freeze and rupture the pipe containing it.

By letter dated August 11, 1994, the licensee for Indian Point Unit 1 submitted its response to the NRC on Bulletin 94-01. The licensee described five key steps in a work plan it developed with the express objective of identifying and quantifying non-evaporative losses of water from the Unit 1 spent fuel pools: (1) isolate the east/west pools containing spent fuel from four smaller pools using new gates, (2) install new water level monitoring instrumentation in the west pool, (3) de-water the four smaller pools, (4) monitor and sample various plant sumps and offsite locations, and (5) perform mass balance inventory calculations to quantify any inventory losses and demonstrate they are being recovered by the plant's subsurface drain system. The licensee also informed the NRC of a "*hydro-geological assessment of the potential for any leakage from the storage pools to affect ground water supplies and the range of such influence were it determined to be capable of occurring.*" The Indian Point Unit 1 FSAR contains no mention of Bulletin 94-01, the gates installed to separate the pools, the new spent fuel pool water level instrumentation, the inventory management program, or hydro-geological assessment results.

CONCLUSIONS

The Indian Point licensees failed to comply with the regulatory requirement in 10 CFR 50.71(e) to update the FSARs to reflect safety analyses performed at the request of the NRC. Even the 1996 reminder from the NRC following the problems at Millstone failed to stop the non-compliance problem and to remedy past shortfalls.

As a direct consequence of violating this regulation, the Indian Point Unit 1, 2, and 3 FSARs do not adequately contain all of the required safety analyses information.

The inadequate FSARs, it is impossible to ascertain the adequacy of the aging management programs for Indian Point. The inadequate FSARs do not fully describe the safety functions performed by structures, systems, and components within the design and licensing bases, making it impossible to first establish that all required structures, systems, and components are properly included within the scope of the aging management programs and then evaluate whether the scope and methodology of the aging management programs for those structures, systems, and components is adequate to provide reasonable assurance that the credited safety functions will be performed.

Prepared by: David A. Lochbaum
Director, Nuclear Safety Project
Union of Concerned Scientists
1707 H Street NW, Suite 600
Washington, DC 20006
(202) 331-5430

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
Generic Letter 82-09, "Environmental Qualification of Safety Related Electrical Equipment," dated 04/20/1982		***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 7.1.4 states that Generic Letter 82-09 accepted the use of peak accident temperature rather than saturation temperature for EQ purposes.***		
Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," dated 06/02/1982		08/02/1982 Response: "...the blanket insulation [on Reactor Coolant Pump 23] was changed to a preformed metal type to preclude potential future build-up [of boric acid]."	08/03/1982 Response: no design basis changes identified.	
Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Emergency Response Capabilities," dated 12/17/1982		04/15/1983 Response: Con Ed described its plans for an SPDS, detailed control room design review, EOPs, emergency response facilities and other TMI-related items. 09/12/1986 RAI Response: Con Ed responded to open items from the NRC's technical evaluation report. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section***	04/18/1983 Response: NYPA committed to install an SPDS 08/23/1983 Response: NYPA committed to develop an SPDS and to install it by specified dates. 04/16/1985 Response: NYPA plans to implement SPDS during the Cycle 4/5 refueling outage. 08/06/1985 Order: NRC ordered NYPA to complete	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		7.1.5 states that IP-2 instrumentation compliance with Reg Guide 1.97 Rev. 2 as required by Generic Letter 82-33 has been addressed in various submittals. *IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 7.7.1 states that a detailed control room design review was conducted in response to Generic Letter 82-33.* *IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 12.3.1 states that the IP-2 procedures generation package (PGP) and emergency operating procedures (EOPs) were developed in accordance with Generic Letter 82-33.* *IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 12.7.2 states that emergency response facilities are*	the Generic Letter 82-33 actions per the schedule in NYPA's 06/29/1984 submittal	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		addressed as required by Generic Letter 82-33.		
Generic Letter 83-35, "Clarification of TMI Action Plan Item II.K.3.31," dated 11/02/1983		08/26/1986 Response: Con Ed cited WCAP-11145 analyses. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 14.3.3.4 states that evaluations in accordance with Generic Letter 83-35 were performed and documented in WCAP-11145.***	11/13/1986 Acceptance: NRC accepted NYPA's 07/09/1985 and 08/18/1986 letters citing WCAP-11145 analyses. ***IP-3 UFSAR submitted with the License Renewal Application Section 14.3.3.3 states that evaluations in accordance with Generic Letter 83-35 were performed and documented in WCAP-11145.***	
Bulletin 83-08, "Electrical Circuit Breakers With Undervoltage Trip...in Safety-Related Applications other than the Reactor Trip System," dated 12/28/1983		04/06/1984 Response: no undervoltage trip attachments used in safety-related applications other than the reactor trip breakers		
Bulletin 84-03, "Refueling Cavity Water Seal," dated 08/24/1984		03/31/1987 Response: IP-2 uses a 4-inch wide, 60 durometer seal as opposed to the 3 1/2 –inch wide, 40 durometer seal at Haddam Neck. Describes test performed showing no	11/27/1984 Response: calculation of consequences from postulated reactor cavity seal failure with operator response times. 03/29/1985 Response:	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		leakage even with seal deflated. Describes a postulated 6-inch pipe break and the associated reactor cavity drain time.	describes existing design features such as refueling machine interlocks that minimize chances of a fuel assembly being dropped onto the reactor cavity seal.	
Generic Letter 85-01, "Fire Protection Policy Steering Committee Report," dated 01/09/1985			01/03/1986 Response: provides evaluation of a fire door frame assembly that is not labeled by an approved independent testing lab. The evaluation includes eight factors (assumptions) needed to support the evaluation's conclusion. ***IP-3 UFSAR submitted with the License Renewal Application Section 9.6.2.2 states that an evaluation for this unrated fire door was performed per Generic Letter 85-01.***	
Generic Letter 85-02, "Staff Recommended Actions ... Regarding Steam Generator Tube Integrity," dated 04/17/1985			07/07/1987 Amendment 76: Revised steam generator tube inspection to reflect GL 85-02.	
Generic Letter 85-12, "Implementation Of TMI Action Item		12/20/1985 Response: One of the two RCP pressure instruments will be replaced	12/13/1985 Response: NYPA provided the plant-specific complement to the generic	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
II.K.3.5, "Automatic Trip Of Reactor Coolant Pumps," dated 06/28/1985		with full-range instrument to meet Reg Guide 1.97. Westinghouse's LOFTRAN code was used to analyze the RCP trip criteria. The EOPs were revised to include the RCP trip response steps.	analysis by the WOG. 11/19/1986 Acceptance: NRC accepted NYPA's reliance on the WOG analysis. ***IP-3 UFSAR submitted with the License Renewal Application Section 14.0.1 states that the reactor coolant pump trip analysis addressed all of the points in Generic Letter 85-12.***	
Bulletin 85-01, "Steam Binding of Auxiliary Feedwater Pumps," dated 10/29/1985			02/18/1986 Response: describes changes to operating procedures and operators rounds	
Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," dated 11/15/1985		01/22/1988 Response: Con Ed outlines plans. 02/26/1988 Response: Con Ed stated that bulletin only applied to 13 valves in the high pressure coolant injection system. Engineering reviews of the thermal overload heater sizing led to replacement of the overload devices on two of the valves.	05/13/1986 Response: NYPA stated that the bulletin only applied to only 10 motor-operated valves in the high head safety injection system and described the existing design features for these valves.	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		05/09/1988 RAI Response: Con Ed addressed NRC's questions about 13 MOVs. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 6.2.2.3.16 states that "in response to the IE Bulletin 85-03, the operability of key safety Motor Operated Valves was verified with associated full differential pressure."***		
Generic Letter 85-22, Potential For Loss Of "Post-LOCA Recirculation Capability Due To Insulation Debris Blockage," dated 12/03/1985			***IP-3 UFSAR submitted with the License Renewal Application Section 6.2.2 states that the actual containment water level prior to swapover to the recirculation phase would be sufficient to provide adequate NSPH for the pumps, based upon a review performed in response to Generic Letter 85-22.***	
Generic Letter 86-10, "Implementation of Fire Protection Requirements," dated 04/24/1986			***IP-3 UFSAR submitted with the License Renewal Application Section 9.6.2.2 states that fire protection features within the turbine building were evaluated per***	

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NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
			Generic Letter 86-10.	
Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment In Operating Reactors (USI A-46)," dated 02/19/1987		10/04/1988 Response: Con Ed committed to comply with the Generic Implementation Procedure (GIP) in its entirety. 09/21/1992 Response: Safe Shutdown Equipment List (SSEL) has been developed and additional walkdowns are required. Con Ed intends to comply with the Generic Implementation Procedure (GIP) in its entirety. "In accordance with 10 CFR 50.71(e), the Updated Final Safety Analysis Report (UFSAR) would consequently be updated." ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 1.11.7 states that the licensee committed to implement the Generic Implementation Procedure (GIP-2) including the clarifications, interpretations, and exceptions in the NRC's Supplemental Safety***	09/22/1992 Response: SSEL will be developed by March 31, 1994, and walkdowns conducted during the refueling outage scheduled for April 1994. 11/16/1995 Response: NYPA transmitted two reports required by the NRC – (1) technical report for SSEL selection and relay/contact evelaution for seismic adequacy, and (2) evaluation report of the seismic adequacy of mechanical and electrical equipment.	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		Evaluation Report (SSER-2).		
Generic Letter 87-11, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," dated 06/19/1987		***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 1.11.3 states that the original design criteria in Table 1.11-2 have been modified to eliminate pipe whip restraints and jet impingement shields per Generic Letter 87-11.***		
Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," dated 07/09/1987		09/11/1987 Response: describes erosion/corrosion inspection procedures for piping ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 10.4 states that an inspection program in response to Bulletin 87-01 has been established and references this NRC submittal.***	09/15/1987 Response: ditto	
Generic Letter 87-12, "Loss of Residual Heat Removal While The Reactor Coolant System is Partially Filled," dated 07/09/1987		09/29/1987 Response: Con Ed described existing procedures and design features for monitoring RCS conditions during shut down and provided the results from a dose consequence	09/21/1987 Response: NYPA described existing procedures and design features for monitoring RCS conditions during shut down. 12/30/1987 Response: NYPA	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		calculation of a postulated loss of RCS decay heat removal accident.	makes minor editorial corrections to its 09/21/1987 and 11/13/1987 responses. ***IP-3 UFSAR submitted with the License Renewal Application Section 4.3.7 states that a new system to monitor reactor coolant system water level during cold shutdown was implemented in response to Generic Letter 87-12.***	
Bulletin 88-01, "Defects in Westinghouse Circuit Breakers," dated 02/05/1988			08/04/1988 Response: provides results of breaker tests	
Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," dated 02/05/1988		11/23/1987 Response (enclosure to 03/25/1988 response): Con Ed provided the results from stability and stress analyses for tubes in steam generators 22 and 24. Con Ed explained the procedure revisions it would make for the response to primary-to-secondary leak indications. 03/25/1988 Response: Con Ed stated procedures were	03/24/1988 Response: NYPA described existing equipment and procedures and committed to replacing the steam generators in the next refueling outage.	Table 3.1.1 Item 3.1.1-79 states that Aging Management Program takes corrective actions consistent with Bulletin 88-02.

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		revised on how primary-to-secondary leaks would be monitored with and without condenser air ejector monitor R-15 available. 01/19/1990 RAI Response: Con Ed submitted copies of WCAP-11811 and WCAP-11812, evaluations of vibration-induced tube fatigue. Two tubes in steam generator 21 plugged based on analysis results. 05/08/1990 NRC Letter: NRC informed Con Ed about information it received from Westinghouse that a re-evaluation was needed for IP-2. 05/11/1990 RAI Response: Con Ed provides a plant-specific Westinghouse request for withholding of their analysis. 10/18/1990 Acceptance: NRC notified Con Ed that its responses had been accepted.		

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		08/10/1994 Response: Con Ed stated an N-16 monitoring system had been installed to monitor for steam generator tube leakage. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Chapter 4 page 52 states that U-bend fatigue is not a consideration for the replacement steam generators due to the stainless steel tube support plates.***		
Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants," dated 03/17/1988		05/31/1988 Response: describes RCS inspection scope and frequency	06/01/1988 Response: Describes commitments to develop list of vulnerable components and develop procedures for appropriate inspections	
Bulletin 88-04, "Potential Safety-Related Pump Loss," dated 05/05/1988			***IP-3 UFSAR submitted with the License Renewal Application Section 6.2.2 states that strong pump / weak pump interactions described in Bulletin 88-04 are prevented by the recirculation line configuration.***	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated 07/12/1988		03/03/1990 Response: transmitted formal evaluation by Southwest Research Institute of reactor vessel material specimen	01/13/1989 Response: revised methodology requires changes before cycle 8	
Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," dated 07/26/1988		04/20/1989 Response: discussion of design and inspection results	04/25/1989 Response: results of thimble tube thickness measurements	
Generic Letter 88-12, "Removal of Fire Protection Requirements from Technical Specification," dated 08/02/1988		06/16/1994 Request: Con Ed submitted a license amendment request to relocate fire protection requirements from the technical specifications per the generic letter. 03/26/1996 Amendment: The NRC issued Amendment 186 which relocated fire protection requirements from the technical specifications per the generic letter.	04/18/1994 Request: NYPA submitted a license amendment request to relocate fire protection requirements from the technical specifications per the generic letter.	
Generic Letter 88-17, "Loss of Decay Heat Removal," dated		02/03/1989 Response: Con Ed described plans to install two separate and diverse	01/03/1989 Response: NYPA described existing equipment and procedures used during	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
10/17/1988		RCS level monitoring instrumentation systems by the end of the 1989 refueling outage. Con Ed also described an indicator being added in the control room to allow the operators to monitor low RHR flow conditions. Con Ed stated that procedures would be revised to de-energize motor-operated valves MOV-730 and MOV-731 when they are opened with RHR in service. 08/22/1990 Response: Con Ed revised its commitment on establishing a vent pathway during RCS draindown. The new commitment is to provide a vent path prior to RCS draindown to the level that RHR vortexing is possible. Con Ed stated an ultrasonic level detector had been installed. 09/20/1991 Response: Con Ed notified NRC that all actions and hardware modifications had been	mid-loop operation. ***IP-3 UFSAR submitted with the License Renewal Application Section 1.3.2 states that two independent reactor coolant system water level indication systems were installed in response to Generic Letter 88-17.*** ***IP-3 UFSAR submitted with the License Renewal Application Section 4.3.7 states that new methods of monitoring reactor coolant system water level during cold shutdown were implemented in response to Generic Letter 88-17.***	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		completed. 07/28/1997 Response: A Reactor Coolant System Redundant Level Measuring System, with an alarm capability, was installed during the 1997 refueling outage.		
Generic Letter 88-18, "Plant Record Storage on Optical Disks," dated 10/20/1988		***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 11.4 states that IP-2 procedures for storing records on optical disks complies with Generic Letter 88-18.***		
BULLETIN 88-11, "Pressurizer Surge Line Thermal Stratification," dated **12/20/1988**		03/03/1989 Response: more time for analysis needed 02/04/1991 Response: more time for analysis needed 10/01/1991 Response: Con Ed provided its closeout package in response to the bulletin, including transmittal and proprietary and non-proprietary versions of the plant-specific analyses performed by Westinghouse.	03/21/1989 Response: more time for analysis needed 05/30/1989 Response: provides justification for interim operation based on WOG evaluation 01/18/1991 Response: more time for analysis needed ***IP-3 UFSAR submitted with the License Renewal Application Section 1.3.4 states that a new***	Section 4.3.1.8 states that IP's original design analyses did not consider thermal stratification of the surge line, but a program was developed for IP to ensure component integrity.

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 4.2.2.7 states that thermal stratification effects on the pressurizer surge lines have been evaluated for the design life of the plant.	***methodology per Generic Letter 88-11 was applied to the evaluation of reactor vessel capsule Z.***	
Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," dated 07/18/1989			***IP-3 UFSAR submitted with the License Renewal Application Section 6.4.2 states that containment fan cooler cooling coil thermal performance is assured by means other than roughing filters and references Generic Letter 89-13.***	
Generic Letter 92-01, "Reactor Vessel Structural Integrity," dated 02/28/1992		07/06/1992 Response: Con Ed stated "This report describes the methods used and results obtained in evaluating Indian Point Unit 2 relative to GL 92-01." ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 4.2.5 states that additional information on nil ductility transition temperatures for pressurized thermal shock in response to Generic Letter***		Table 4.2-4 9th column is "Chemistry Factor GL 92-01"

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		92-01, Rev. 1, is provided in a reference.		
Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54(f)," dated 06/21/1993		12/11/1997 Response: Con Ed notified NRC that the Salem rod withdrawal modification had been completed during the 1997 refueling outage. 05/04/1998 Response: Con Ed notified NRC that the modification and testing recommended by Westinghouse had been completed during the 1997 refueling outage. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 3.2.3.4.1.7 states that the sequence timing has been modified to preclude the rod withdrawal event described in Generic Letter 93-04.***	06/11/1998 Response: NYPA provided confirmation that actions taken in response to the generic letter had been completed. NYPA stated that the post-modification testing was completed.	
Bulletin 94-01, "Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1," dated 04/14/1994	08/11/1994 Response: describes plans to install new level instrumentation in the west pool and various	N/A	N/A	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
	administrative controls. Describes two new gates already installed to separate the spent fuel pools. ***The IP-1 FSAR submitted with the License Renewal Application contains no mention of Bulletin 94-01, the licensee's response, the gates installed to separate the spent fuel pools, or the new level instrumentation in the west pool.***			
Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves," dated 08/17/1995		***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 6.1.1.5 states that evaluations of safety-related power-operated gate valves were performed in response to Generic Letter 95-07.***		
Generic Letter 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," dated		10/23/1996 Response: describes monitoring program established after boraflex panels installed in	08/06/1996 Response: boraflex not used at IP-3	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
06/26/1996		1990		
Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated 09/30/1996		01/28/1997 Response: describes analysis showing containment response to postulated event 08/29/1997 Response: addresses "the corrective actions and modifications made during the 1997 refueling outage, which restored certain lines at Indian Point Unit No. 2 to full compliance with Generic Letter 96-06." Ex: "The original piping insulation installed on line #10 was changed from two inches of calcium silicate to six inches of glass wool" and "Line #69, 26 had relief valves installed." 09/15/1998 Response: reply to NRC RAI on analytical methodology and assumptions. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 6.1.1.5 states that isolated***	10/14/1997 Response: "Based on evaluations performed for the service water supply and return piping system for postulated waterhammer loading as outlined in NRC GL 96-06, a total of ten (1) supports were modified during R09 to provide additional support capabilities and minor function change." 10/23/1998 Response: reply to NRC RAI on analytical methodology and assumptions	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
		piping segments penetrating the containment were evaluated in response to Generic Letter 96-06. ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 9.6.1.3 states that the containment fan cooler units were evaluated for waterhammer potential in response to Generic Letter 96-06.***		
Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated 04/01/1997		05/01/1997 Response: schedule info only 07/30/1997 Response: no design basis changes identified 01/12/1999 Response: provided EPRI evaluation	07/21/1997 Response: schedule info only 02/16/1999 Response: referenced EPRI evaluation	
Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of		11/10/1998 Response: design basis for containment coatings ***IP-2 UFSAR Rev. 20 submitted with the License Renewal Application Section 6C.6 references Generic Letter 98-04.***	11/10/1998 Response: design basis for containment coatings	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated 07/14/1998				
Generic Letter 99-02, "Laboratory Testing of Nuclear- Grade Activated Charcoal," dated 03/06/1992			08/02/1999 Response: NYPA committed to testing per ASTM D3803-1989 and to revising technical specifications accordingly. ***IP-3 UFSAR submitted with the License Renewal Application Section 1.3.2 states that nuclear grade activated charcoal is tested per response to Generic Letter 99-02.*** ***IP-3 UFSAR submitted with the License Renewal Application Section 6.4.2 states that activated carbon is tested per ASTM D3803-1989 per response to Generic Letter 99-02.*** ***IP-3 UFSAR submitted with the License Renewal Application Appendix 6C states that activated charcoal***	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
			is tested per response to Generic Letter 99-02.	
Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," dated 06/09/2003		08/07/2003 Response: describes design	08/07/2003 Response: describes design	
Generic Letter 2003-01, "Control Room Habitability," dated 06/12/2003		06/28/2005 Response: "Entergy has completed these steps … Control Room Ventilation System flow path and operating configuration modification"	08/06/2003 Response: more time needed for analysis 10/26/2004 Response: proposed license amendment to address GL concerns 06/28/2005 Response: "Entergy has completed these steps … Control Room Ventilation System flowpath and operating configuration modification"	
Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated 09/13/2004		02/28/2005 Response: schedule plans 09/01/2005 Response: plans for calculations and modifications 12/15/2005 Response: schedule commitments	02/28/2005 Response: schedule plans 09/01/2005 Response: plans for calculations and modifications 12/15/2005 Response: schedule commitments	

Table 1
NRC Generic Correspondence Cited in Indian Point FSARs and/or License Renewal Application

NRC Generic Correspondence	Indian Point 1	Indian Point 2	Indian Point 3	License Renewal Application
Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated 02/01/2006		04/03/2006 Response: no design basis changes identified 01/31/2007 RAI Response: no design basis changes identified	04/03/2006 Response: no design basis changes identified 01/31/2007 RAI Response: no design basis changes identified	
Generic Letter 2007-01, **title?** 02/07/2007		05/07/2007 Response: operational experience reported	05/07/2007 Response: operational experience reported	

David A. Lochbaum

EDUCATION

June 1979 Bachelor of Science in Nuclear Engineering, The University of Tennessee at Knoxville 37996

June 1976 Diploma, Farragut High School, Knoxville, TN 37934

Note: I received a diploma from the Farragut High School although I did not attend a single day of class there. After finishing my junior year at the Terry Parker High School in Jacksonville, FL 32211, my family moved to Knoxville. With the advanced courses taken at Terry Parker and high score on the ACT test, I skipped my senior year of high school to enroll at The University of Tennessee in September 1975.

EXPERIENCE SUMMARY

10/96 to date *Director - Nuclear Safety Project*
Union of Concerned Scientists
Two Brattle Square
Cambridge, MA 02238-9105
Supervisors: Lisbeth Gronlund and David Wright, Co-Directors of the Global Security Program at (617) 547-5552

Responsible for directing UCS's nuclear safety program, for monitoring developments in the nuclear industry, for serving as the organization's spokesperson on nuclear safety issues, for initiating action to correct safety concerns, for authoring reports and briefs on safety issues, and for presenting findings to the Nuclear Regulatory Commission, the US Congress, and state and local officials.

11/87 to 09/96 *Senior Consultant*
Enercon Services, Inc.
500 Townpark Lane, Suite 275
Kennesaw, GA 30144-5509
Supervisor: Carter Noland at (770) 919-1930

Responsible for developing the conceptual design package for the alternate decay heat removal system, for closing out partially implemented modifications, reducing the backlog of engineering items, and providing training on design and licensing bases issues at the Perry Nuclear Power Plant.

Responsible for developing a topical report on the station blackout licensing bases for the Connecticut Yankee plant.

Responsible for vertical slice assessment of the spent fuel pit cooling system and for confirmation of licensing commitment implementation at the Salem Generating Station.

Responsible for developing the primary containment isolation devices design basis document, reviewing the emergency diesel generators design basis document, resolving design document open items, and updating design basis documents for the FitzPatrick Nuclear Power Plant.

Responsible for the design review of balance of plant systems and generating engineering calculations to support the Power Uprate Program for the Susquehanna Steam Electric Station.

Responsible for developing the reactor engineer training program, revising reactor engineering technical and surveillance procedures and providing power manuevering recommendations at the Hope Creek Generating Station.

Responsible for supporting the lead BWR/6 Technical Specification Improvement Program and preparing licensing submittals for the Grand Gulf Nuclear Station.

03/87 to 08/87 *System Engineer*
General Technical Services
Columbia, MD (company no longer in business)
Supervisor: James Bleier

Responsible for reviewing the design of the condensate, feedwater and raw service systems for safe shutdown and restart capabilities at the Browns Ferry Nuclear Plant.

08/83 to 02/87 *Senior Engineer*
Enercon Services, Inc.
500 Townpark Lane, Suite 275
Kennesaw, GA 30144-5509
Supervisor: Carter Noland at (770) 919-1930

Responsible for performing startup and surveillance testing, developing core monitoring software, developing the reactor engineer training program, and supervising the reactor engineers and Shift Technical Advisors at the Grand Gulf Nuclear Station.

10/81 to 08/83 *Reactor Engineer / Shift Technical Advisor*
Tennessee Valley Authority
Browns Ferry Nuclear Plant
Athens, AL 35611
Supervisor: Earl Nave (now retired)

Responsible for performing core management functions, administering the nuclear engineer training program, maintaining ASME Section XI program for the core spray and CRD systems, and covering STA shifts at the Browns Ferry Nuclear Plant.

06/81 to 10/81 *BWR Instructor*
General Electric Company
BWR/6 Training Center
Inola, OK (site no longer in business)

Responsible for developing administrative procedures for the Independent Safety Engineering Group (ISEG) at the Grand Gulf Nuclear Station.

01/80 to 06/81 *Reactor Engineer / Shift Technical Advisor*
Tennessee Valley Authority
Browns Ferry Nuclear Plant
Athens, AL 35611
Supervisor: Earl Nave (now retired)

Responsible for directing refueling floor activities, performing core management functions, maintaining ASME Section XI program for the RHR system, providing power maneuvering recommendations and covering STA shifts at the Browns Ferry Nuclear Plant.

06/79 to 12/79 *Junior Engineer*
Georgia Power Company
Edwin I. Hatch Nuclear Plant
Baxley, GA 31513
Supervisor: Steve Curtis

Responsible for completing pre-operational testing of the radwaste solidification systems and developing design change packages for modifications to the liquid radwaste systems at the Edwin I. Hatch Nuclear Plant.

OTHER QUALIFICATIONS

April 1982	Training culminating in certification as a Shift Technical Advisor at the TVA Browns Ferry Nuclear Plant
1981	Received Best Idea Award at Browns Ferry for change to control rod drive friction testing procedure that significantly reduced personnel radiation exposures
May 1980	Training culminating in certification as an Interim Shift Technical Advisor at the TVA Browns Ferry Nuclear Plant

Member, American Nuclear Society (since 1978).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Operations, Inc.

Docket Nos. 50-247-LR and 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

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DECLARATION OF PAUL BLANCH

Paul Blanch, hereby declares under penalty of perjury that the following is true and correct:

1. I have been retained by the New York State Office of the Attorney General to provide expert services in connection with the application by Entergy Nuclear Operations, Inc. and its affiliates (collectively Entergy) for a renewal of the two separate operating licenses for the nuclear power generating facilities located at Indian Point.

2. Beginning in 1964, I served in the U.S. Navy as a nuclear reactor and electric plant operator on *Polaris* class submarines for seven years. As part of my Navy duties, I was certified as an instructor at the Navy prototype reactor (S1C) in Windsor Locks, Connecticut. Thereafter, in 1972, I received a Bachelor of Science in Electrical Engineering from the University of Hartford. I have more than 25 years of engineering, engineering management, and project coordination experience for the construction and operation of nuclear power plants.

3. I have reviewed the April 30, 2007 License Renewal Application submitted by Entergy to renew the operating licenses for Indian Point Unit 2 and Unit 3. As set forth below and as developed in the relevant Contentions contained in the Petition to Intervene of the State of New York, it is my opinion that the proposed aging management programs fail to provide reasonable assurance that IP2 and IP3 will operate safely through their proposed license renewal periods.

4. Failure to properly manage aging of Non-environmentally-qualified (Non-EQ) Inaccessible Medium-Voltage Cables may challenge:

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.

5. The failure to properly manage aging of the Non-EQ Inaccessible Medium-Voltage Cables could result in the loss of the 6.9 kV and 13.8 kV safety related buses that supply emergency power to the 480 volt safety equipment including Station Blackout (SBO) loads, service water motors/pumps, safety injection pumps, and other electrical loads required to meet the requirements of 10 C.F.R. §§ 54.4 and 54.29.

6. Consequence of failures of Non-EQ Inaccessible Medium-Voltage Cables may result in accidents beyond the Design Basis Accidents resulting in exposures to the public exceeding 10 C.F.R. § 100 limits.

7. The applicant has not "demonstrate(d) 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" (10 C.F.R. § 54.21(a)(3)) for those SSC's identified for Pressurized Water Reactors in Table 6 of NUREG 1801.

8. The applicant has failed to identify the location and extent of Non-EQ Inaccessible Medium-Voltage Cables in use at IP 2 and IP 3. For example, the Applicant has failed to provide drawings referenced in the LRA. *See* the reference to Drawing 9321-F-31193.

9. The applicant has failed to provide access to referenced documents that are not publicly available (e.g. EPRI TR-103834-P1-2 and EPRI TR-109619). A computer search has been conducted by me of all publicly available documents using ADAMS, CITRIX, BRS, GOOGLE and the EPRI web site and the search has not located these referenced documents. It is not possible to fully evaluate the adequacy of the AMP without these references.

10. The applicant has failed to provide a copy of its "Non-EQ Insulated Cables And Connections Program" and "Non-EQ Inaccessible Medium-voltage Cable Program" identified in Appendix B of the LRA.

11. The applicant has failed to address specific recommendations from the referenced Sandia report (SAND96-0344).

12. There is no technical basis to support life extension using the existing medium voltage power cables without a descriptive aging management plan.

13. There is no technical basis to justify differences between programs for aging management of accessible cables and inaccessible cables. 10 C.F.R. § 54.21(a)(3).

14. A review of all documents supplied as part of the LRA has failed to identify which cables are encompassed by the AMP. A review of the "one line" electrical drawing from Chapter

8 of the IP 2 UFSAR confirms that many of these medium voltage cables are within the scope of 10 C.F.R. § 54.4.

15. The applicant has failed to provide a copy of its "NonEQ Insulated Cables and Connections Program." It is not possible to assess the adequacy of the AMP without a copy of this program as described in LRA B.1.25. No details are provided explaining the Non-EQ Inaccessible Medium-Voltage Cable Program except that it appears to be limited to ". . . inspections for water accumulation in manholes at least once every two years." *Id.* Experience indicates that not all inaccessible cables are capable of inspection via "manholes".

16. The only difference between the cables discussed in B.1.23 and B.1.25 is accessibility which, in light of the comparable safety significance of both types of cables and the risk of aging damage to both types of cables is not a technically defensible basis for treating the two types of Non-EQ cables differently.

17. There are numerous inaccessible cables (less than 2 kV) ranging in voltage from 100 to 2,000 volts installed at the IP 2 and IP 3 that meet the requirement as described in 10 CFR § 54.4 including power and control for the following vital components.

- Auxiliary component cooling pumps
- Safety injection pumps
- Residual heat removal pumps
- Nuclear service water pumps
- Containment air recirculation cooling fans
- Auxiliary feedwater pumps
- Spray pumps (if start signal present)

18. The LRA has not specifically identified an aging management program and/or the locations of the Non-EQ Inaccessible Low-Voltage Cables however these cables exist in many locations including power to the Service Water Pumps.

19. The most recent UFSAR confirms the use of these vital cables at IP2 and IP3.

20. Failure to properly manage aging of Non-EQ Inaccessible Low-Voltage Cables may adversely impact:

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. May result in the loss of emergency power to the 480 volt safety equipment including all Station Blackout (SBO) loads.

21. There are numerous Electrical Transformers that perform a function described in §§ 54.4(a)(1)/(2) and (3). Transformers function without moving parts or without a change in configuration or properties as defined in that regulation.

22. Failure to properly manage aging of Electrical Transformers may compromise:

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. 10 C.F.R. §§ 54.4(a)(1)(2) and (3).

23. The consequence of failures of Electrical Transformers may result in accidents beyond the Design Basis Accidents resulting in exposures to the public exceeding 10 C.F.R. § 100 limits.

24. Failure to properly manage aging of electrical transformers could result in loss of emergency power to the 480 volt safety equipment and 6.9kV busses including station blackout loads. Appendix A, Page A-35 of the UFSAR supplement describes a Structures Monitoring Program that includes a program for monitoring "transformer/switchyard support structures" yet there is no APM described for transformers within the scope of 10 C.F.R. § 54.21(a)(1)(i).

25. Many of the legally relevant GDC for IP2 and IP3 relate to components, equipment, and systems that may require aging management. *See e.g.* GDC 47 (Testing of Emergency Core Cooling Systems (Category A); GDC 34 (Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A); GDC 50 (NDT Requirement for Containment Material (Category A)). However, whether aging management is required for those components, equipment and systems cannot be determined nor can the adequacy of any aging management analysis or plan be evaluated until Applicant identifies components, equipment and systems that are in compliance with the legally relevant GDC.

26. There are substantial substantive differences between the trade association version of the GDC and the officially promulgated 1967 Draft GDC. The following are examples of the conflicts:

A. **Criterion 50 from 32 FR 10213 states**: Criterion 50-NDT Requirement for Containment Material (Category A) Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30 degrees F above nil ductility transition (NDT) temperature.

Chapter 5, Page 4 of 89 Revision 20 (541/1698) of the Indian Point 2 UFSAR provided as part of the LRA states: 5.1.1.1.7 Nil-ductility Transition Temperature Requirement for Containment Material - Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50).

Chapter 5, Page 5 of 188 (826/2108) of the Indian Point 3 UFSAR provided as part of the LRA states: Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50 of 7/11/67)

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Both IP2 and IP3 state compliance with GDC 50. However, both UFSARS have reworded and changed the intent of this GDC 50 by removing the words "Principal load carrying components" and "less than 30 degrees F above nil ductility transition (NDT) temperature" from the regulation.

B. **Criterion 47 from 32 FR 10213 states**: Criterion 47-Testing of Emergency Core Cooling Systems (Category A). A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Chapter 6, Page 8 of 120 Revision 20 (Page 717/1698) of the Indian Point 2 UFSAR provided as part of the LRA states: 6.2.1.4 Testing of Emergency Core Cooling System Criterion: 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. (GDC 47)

Chapter 6, Page 10 of 215 (Page 1019/2108) of the Indian Point 3 UFSAR provided as part of the LRA states: Testing of Emergency Core Cooling System Criterion 47: 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.

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Both IP2 and IP3 state compliance with GDC 47. However, the USFARs have reworded and changed the intent of GDC 47 by removing the words "test periodically the delivery capability." The "delivery capability" of the Emergency

Core Cooling System (ECCS) may be impacted by aging mechanisms such as pipe fouling, erosion, corrosion and heat exchanger tube fouling. The License Renewal Application (LRA) has failed to discuss any Aging Management Program (AMP) to assure that the "delivery capability" of the Emergency Core Cooling System (ECCS) continues to meet the requirements of this GDC.

C. **Criterion 34 from 32 FR 10213 states**: Criterion 34-Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A). The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Chapter 4, Page 6 of 85 Revision 20 (Page 443/1698) of the Indian Point 2 UFSAR provided as part of the LRA states: 4.1.3.4 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is 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. (GDC 34)

Indian Point 3 UFSAR provided as part of the LRA fails to address Criterion 34 from the 1967 GDCs.

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IP3 makes no commitment to comply with this regulation. IP2 has completely altered the words and the intent of this General Design Criterion by eliminating the following requirements that may be impacted by aging mechanisms within the scope of 10 CFR 54.4:

(a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve,

(b) to the state of stress of materials under static and transient loadings

(c) to the quality control specified for materials and component fabrication to limit flaw sizes, and

(d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

27. If it is not known whether the components, equipment and systems at IP2 and IP3 are in compliance with the legally relevant GDC it is to be expected that the NRC Staff, in fulfillment of its regulatory obligations, will uncover the deficiencies and require that they be corrected. However, that has not occurred to date and until that has occurred, the specific components, equipment and systems are not known and it is not possible to design or evaluate the appropriate aging management programs and analyses.

28. There are substantial substantive differences between the trade association version of the GDC and the officially promulgated 1967 Draft GDC. The side by side comparison of the two versions is shown in a Chart prepared by me which is attached.

29. Throughout the UFSAR when the language of a GDC with which IP2 and IP3 is alleged to be in compliance is cited, the language is taken from the trade association version of the GDC and not the 1967 Draft GDC.

30. While in a few instances the differences are of little obvious safety significance, in a number of instances the differences are substantial and result in IP2 and the trade association and IP3 and the trade association illegally "granting" IP2 and IP3 an "exemption" from the applicable safety requirements of the AEC. These substantial differences are highlighted on attached Exhibit, prepared by me.

31. Attached to this Declaration is the Chart prepared by me and a copy of my current CV. Both of these documents were prepared by me and are true and correct to the best of my

personal knowledge.

Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated: November 18, 2007
West Hartford, Connecticut

Paul Blanch

Paul Blanch

Comparision of Published (32 FR 10213) General Design Criteria with stated criteria contained within the Indian Point UFSARs

Draft General Design Criteria 1967 Scanned from GDC's as published in 1967 (32 FR 10213)	Indian Point Unit 2 Stated Compliance from the USFAR submitted with LRA	Indian Point Unit 3 Stated Compliance from the USFAR submitted with LRA
PART 50 - LICENSING OF PRODUCTION AND UTILIZATION FACILITIES Introduction. Every applicant for a construction permit is required by the provisions of 50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors. Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required In the In-terest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design In the Interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Cri-teria should be justified. The criteria we designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive Information is needed at the construction permit stage for the items listed in Category A than for those In Category B.	1.3 GENERAL DESIGN CRITERIA (GDC) The General Design Criteria define or describe safety objectives and approaches incorporated in the design of this plant. These General Design Criteria, tabulated explicitly in the pertinent systems sections in this report, comprised the proposed Atomic Industrial Forum versions of the criteria issued for comment by the AEC on July 11, 1967. Also included in this section, are brief descriptions of related plant features, which are provided to meet the design objectives reflected in the criteria at the time of the initial license application. The descriptions are more fully developed in those succeeding sections of the report indicated by the references. More recently, Con Edison completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of Indian Point Unit 2 compliance with the then current General Design Criteria established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to the NRC by Con Edison on August 11, 1980 (Reference 1). Commission concurrence was received on January 19, 1982. The parenthetical numbers following the section headings indicate the numbers of their related proposed Atomic Industrial Forum versions of the General Design Criteria as described in the first paragraph of this section. IP2 FSAR UPDATE Chapter 1, Page 8 of 72 Revision 20, 2006	1.3 GENERAL DESIGN CRITERIA The General Design Criteria establish the necessary design, fabrication, construction, testing and performance requirements for structures, systems, and components important to safety; that is structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants. The General Design Criteria presented and discussed in specific sections of the FSAR (which describe systems, structures, equipment and components important to safety) are those which were in effect at the time when Indian Point 3 was designed and constructed. The General Design Criteria which formed the bases for the Indian Point 3 design were published by the Atomic Energy Commission in the Federal Register of July 11, 1967 and subsequently made part of 10 CFR 50. The Authority completed a study of the method by which the Indian Point 3 facility complied with the safety rules and regulations, in particular those contained in 10 CFR Parts 20 and 50, that were in effect at the time of the study. The study was conducted in accordance with the provisions of NRC Confirmatory Order of February 11, 1980 and were submitted to the NRC on August 11, 1980. The NRC audit of submittal indicated that the Indian Point 3 design and operation meet the applicable regulations. The following sections provide the results of the compliance study, updated to reflect changes made to the configuration since the study was completed. IP3 FSAR UPDATE 4.1.2 General Design Criteria General design criteria which apply to the Reactor Coolant System are given below. The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1976, (sic) and subsequently made a part of 10 CFR 50. The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980 and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.
Criterion 1--Quality Standards (Category A). Those systems and components of reactor facilities which are essential to the pre-vention of accidents which could affect the public health and safety or to mitigation of their consequences 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 or stand-ards on 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 neces-sary. Quality assurance programs, test proce-dures, and Inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, stand-ards, quality assurance programs, test proce-dures, and inspection acceptance levels used is required.	4.1.2.1 Quality Standards 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. (GDC 1)	Quality Standards and Records (Criterion 1) Criterion: Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power plant licensee throughout the life of the unit. 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. (GDC 1 of 7/11/67)
Criterion 2-Performance Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitiga-tion of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenom-ena such as earthquakes, tornadoes, flood-ing conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) Appropriate consideration of the most severe of these natural phenom-ena that have been recorded for the site and the surrounding area and (b) an ap-propriate margin for withstanding forces greater than those recorded to reflect un-certainties about the historical data and their suitability as a basis for design.	4.1.2.2 Performance Standards Criterion: 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. (GDC 2)	Criterion 2: Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed. 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. (GDC 2 of 7/11/67)
Criterion 3-Fire Protection (Category A) The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas contain-ing critical portions of the facility such as containment, control room, and components of engineered safety features.	5.1.1.1.3 Fire Protection Criterion: A reactor facility shall be designed to ensure 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.(GDC 3)	Fire Protection (Criterion 3) Criterion: Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Text in red illustrates some of the differences between published criteria (32 FR 10213) and restated criterion from LRA

Comparision of Published (32 FR 10213) General Design Criteria with stated criteria contained within the Indian Point UFSARs

Draft General Design Criteria 1967 Scanned from GDC's as published in 1967 (32 FR 10213)	Indian Point Unit 2 Stated Compliance from the USFAR submitted with LRA	Indian Point Unit 3 Stated Compliance from the USFAR submitted with LRA
Criterion 4-Sharing of Systems (Category A). Reactor facilities shall not share systems or components unless it is shown safety is not Impaired by the sharing.	6.1.1.7 Sharing of Systems Criterion: 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. (GDC 4)	Environmental and Missile Design Bases (Criterion 4) Criterion: Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.
Criterion 5-Records Requirements (Category A). Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.	4.1.2.3 Records Requirements Criterion: 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. (GDC 5)	Sharing of Structures, Systems and Components (Criterion 5) Criterion: Structures, systems and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.. 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. (GDC 5 of 7/11/67)
Criterion 6-Reactor Core Design (Category A). The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all ex-pected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anti-cipated, including the effects of the low of power to recirculation pumps, tripping out of a turbine generator set. Isolation of the reactor from its primary heat sink, and loss of all off site power.	3.1.2.1 Reactor Core Design Criterion: 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. (GDC 6)	Reactor Core Design Criterion 6: The reactor 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.
Criterion 7-Supression of Power Oscillations (Category B). The core design, together with reliable controls, shall ensure that power oscillations which could cause dam-age In excess of acceptable fuel damage limits are not possible or can be readily suppressed.	3.1.2.2 Suppression of Power Oscillations Criterion: 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. (GDC 7)	Suppression of Reactor Power Oscillations Criterion 7: 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.
Criterion 8--Overall Power Coefficient (Category B).The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.	Compliance not addressed	Compliance not addressed
Criterion 9-Reactor Coolant Pressure Boundary (Category A) The reactor coolant pressure boundary shall be designed and so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.	4.1.3.1 Reactor Coolant Pressure Boundary Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture of significant uncontrolled leakage throughout its design lifetime. (GDC 9)	Criterion: 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. (GDC 9 of 7/11/67)
Criterion 10--Containment (Category A). Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.	5.1.1.1.5 Reactor Containment Criterion: 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. (GDC 10).	Criterion: 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. (GDC 10 of 7/11/67) 1.3.2 Protection by Multiple Fission Product Barriers (Criteria 10 to 19) Reactor Design (Criterion 10) Criterion: The reactor core and associated coolant, control, and protection systems shall 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.
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 access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down, and maintain it in a safe condition if access to the control room is lost due to fire or other cause.	7.2.1.1 Control Room Criterion: 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. (GDC 11)	Reactor Inherent Protection (Criterion 11) Criterion: The reactor core and associated coolant system shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity. 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 exposure of personnel. (GDC 11 of 7/11/67)
Criterion 12-Instrumentation and Control Systems (Category B). Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.	7.1.1 Instrumentation and Control Systems Criteria Criterion: Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables. (GDC 12)	Suppression of Reactor Power Oscillations (Criterion 12) Criterion: The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be Instrumentation and Control Systems Criterion: Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables. (GDC 12 of 7/11/67)

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with stated criteria contained within the Indian Point UFSARs

Draft General Design Criteria 1967 Scanned from GDC's as published in 1967 (32 FR 10213)	Indian Point Unit 2 Stated Compliance from the USFAR submitted with LRA	Indian Point Unit 3 Stated Compliance from the USFAR submitted with LRA
Criterion 13-Fission Process Monitors and Controls (Category B). Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.	7.4.1.1 Fission Process Monitors and Controls Criterion: 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. (GDC 13)	Instrumentation and Control (Criterion 13) Criterion: Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can effect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges Fission Process Monitors and Controls Criterion: 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. (GDC 13 of 7/11/67)
Criterion 14—Core Protection Systems (Category B). Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.	7.2.1.2 Reactor Protection System Criterion: 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. (GDC 14)	Reactor Coolant Pressure Boundary (Criterion 14) Criterion: The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture Core Protection Systems Criterion: 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. (GDC 14 of 7/11/67)
Criterion 15-Engineered Safety Features Protection Systems (Category B). Protection systems. shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.	7.2.1.3 Engineered Safety Features Protection System Criterion: Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features. (GDC 15)	Reactor Coolant System Design (Criterion 15) Criterion: The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Engineered Safety Features Protection Systems Criterion: Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features (GDC 15 of 7/11/67).
Criterion 16-Monitoring Reactor Coolant Pressure Boundary (Category B). Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.	4.1.3.2 Monitoring Reactor Coolant Leakage Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)	Monitoring Reactor Coolant Leakage Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16 of 7/11/67) Containment Design (Criterion 16) Criterion: Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.
Criterion 17-Monitoring Radioactivity Releases (Category B). Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.	6.7.1.1.2 Monitoring Radioactivity Releases Criterion: 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. (GDC 17)	Electrical Power Systems (Criterion 17) Criterion: An onsite electric power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that: 1) Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences. 2) The core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The onsite electrical power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limit and design limit and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a Loss-of-Coolant Accident to assure that the core cooling, containment integrity, and other vital safety functions are maintained. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. Monitoring Radioactivity Releases Criteria: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released form normal conditions, 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 (GDC 17 of 7/11/67).

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Criterion 18 Monitoring Fuel and Waste Storage (Category B). Monitoring and alarm Instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.	11.2.1.2 Monitoring Fuel and Waste Storage Criterion: 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. (GDC 18) 8.1.2.2 10 CFR 50 Appendix A General Design Criterion 18 - Inspection and Testing of Electric Power Systems Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.	Inspection and Testing of Electrical Power System (Criterion 18) Criterion: Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important area and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically: a) The operability and functional performance of the components of the systems such as onsite power sources, relays, switches, and buses. b) The operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system and the onsite power system.
Criterion 19-Protection Systems Reliability (Category B). Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.	7.2.1.4 Protection Systems Reliability Criterion: 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. (GDC 19) 9.4.1.6 Codes and Standards System code requirements are given in Table 9.4-1. In addition, the high radiation sampling system was designed and installed to meet the provisions of NUREG-0737. These provisions include the following: 1. Provide postaccident sampling and analysis capability. The combined time for sampling and analysis is 3 hr or less from the time a decision is made to take a sample. 2. Provide capability to obtain and analyze a sample without radiation exposure to any individual exceeding the criteria of GDC 19 (10 CFR Part 50, Appendix A).	Control Room (Criterion 19) Criterion: A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 Rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
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 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. Different principles shall be used where necessary to achieve true independ-ence of redundant instrumentation com-ponents.	7.2.1.5 Protection Systems Redundancy and Independence Criterion: 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 function to be served. (GDC 20)	Protection Systems Redundancy and Independence Criterion: Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure on 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 function to be served. (GDC 20 of 7/11/67) Protection System Functions (Criterion 20) Criterion: The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems to assure that specified acceptable Fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important
Criterion 21-Single Failure Definition (Category B). Multiple failures resulting from a single event shall be treated as a single failure.	Compliance not addressed.	Protection System Reliability and Testability (Criterion 21) Criterion: The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that
Criterion 22-Separation of Protection and Control Instrumentation Systems (Category B). Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.	Compliance not addressed.	Protection System Independence (Criterion 22) Criterion: The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

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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.	7.2.1.6 Protection Against Multiple Disability for Protection Systems Criterion: 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. (GDC 23)	Protection Against Multiple Disability for Protection Systems Criterion: 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 basis. (GDC 23 of 7/11/67) Protection System Failure Modes (Criterion 23) Criterion: The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, pain, and radiation) are experienced.
Criterion 24-Emergency Power for Protection Systems (Category B). In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.	8.1.1.2 Emergency Power Criterion: 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 component. (GDC 39 and GDC 24)	Separation of Protection and Control Systems (Criterion 24) Criterion: The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired. Emergency Power Criterion: 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 component. (GDC 39 and GDC 24 of 7/11/67)
Criterion 25–Demonstration of Functional Operability of Protection Systems (Category B). Means shall be included for testing protection systems while the reactor is in opera-tion to demonstrate that no failure or loss of redundancy has occurred.	7.2.1.7 Demonstration of Functional Operability of Protection Systems Criterion: 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. (GDC 25)	Protection System Requirements for Reactivity Control Malfunctions (Criterion 25) Criterion: The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods. Demonstration of Functional Operability of Protection Systems Criterion: 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. (GDC 25 of 7/11/67)
Criterion 26-Protection Systems Fail-Safe Design (Category B). The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as discon-nection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.	7.2.1.8 Protection Systems Failure Analysis Design Criterion: The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced. (GDC 26)	Reactivity Control System Redundancy and Capability (Criterion 26) Criterion: Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions. Protection System Failure Analysis Design Criterion: The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced. (GDC 26 of 7/11/67)
Criterion 27-Redundancy of Reactivity Control (Category A). At least two independent reactivity control systems, preferably of different principles, shall be provided.	3.1.2.3 Redundancy of Reactivity Control Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27)	Redundancy of Reactivity Control Criterion: Two independent control systems, preferably of different principles, shall be o provided. (GDC 27 of 7/11/67) Combined Reactivity Control System Capability (Criterion 27) Criterion: The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained

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Criterion 28-Reactivity Hot Shutdown Capability (Category A). At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.	**3.1.2.4 Reactivity Hot Shutdown Capability** **Criterion: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. (GDC 28)**	**Reactivity Limits (Criterion 28)** Criterion: The reactivity control system shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition. **Reactivity Hot Shutdown Capability** Criterion 28: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. **Reactivity Hot Shutdown Capability** Criterion: The reactivity control system provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. (GDC 28 of 7/11/67)
Criterion 29-Reactivity Shutdown Capability (Category A). At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.	**3.1.2.5 Reactivity Shutdown Capability** **Criterion: 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	**Protection Against Anticipated Operational Occurrence (Criterion 29)** Criterion: The protection and reactivity control systems shall be designed to assume an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. **Reactivity Shutdown Capability** Criterion: 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. (GDC 29 of 7/11/67)
Criterion 30-Reactivity Holddown Capability (Category B). At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with, appropriate margins for contingencies.	**3.1.2.6 Reactivity Holddown Capability** **Criterion: 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. (GDC 30)**	**Reactivity Hold-Down Capability** Criterion: 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 to the public. (GDC 30 of 7/11/67) **Quality of Reactor Coolant Pressure Boundary (Criterion 30)** Criterion: Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. **Reactivity Holddown Capability** Criterion 30: 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 that there will be no undue risk to the health and safety of the public.
Criterion 31-Reactivity Control Systems Malfunction (Category B). The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.	**3.1.2.7 Reactivity Control Systems Malfunction** **Criterion: 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. (GDC 31)**	**Fracture Prevention of Reactor Coolant Pressure Boundary (Criterion 31)** Criterion: The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws. **Reactivity Control System Malfunction** Criterion 31: 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, buy limiting reactivity transients to avoid exceeding acceptable fuel damage limits. **Reactivity Control Systems Malfunction** Criterion: 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. (GDC 31 of 7/11/67)

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Criterion 32-Maximum Reactivity Worth of Control Rods (Category A). Limits, which include considerable 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 impair the effectiveness of emergency core cooling.	3.1.2.8 Maximum Reactivity Worth of Control Rods Criterion: 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. (GDC 32)	Inspection of Reactor Coolant Pressure Boundary (Criterion 32) Criterion: Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an Maximum Reactivity Worth of Control Rods Criterion 32: 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.
Criterion 33-Reactor Coolant Pressure Boundary Capability (Category A). The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any in-advertent 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 inser-tion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.	4.1.3.3 Reactor Coolant Pressure Boundary Capability Criterion: 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. (GDC 33)	Reactor Coolant Makeup (Criterion 33) Criterion: A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.
Criterion 34-Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A). The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for con-trol over service temperature and irradiation effects which may require operational restrictions.	4.1.3.4 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is 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. (GDC 34)	Residual Heat Removal (Criterion 34) Criterion: A system to remove residual heat shall be provided. The safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capability 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 not available), the system safety function can be accomplished,
Criterion 35-Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A). Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be sub-jected to potential loadings, such as a reactivity induced loading, service tempera-tures shall be at least 120° F. above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60° F. above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy	Compliance not Addressed	Compliance not Addressed
Criterion 36-Reactor Coolant Pressure Boundary Surveillance (Category A). Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance 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 ASTM-E-185-M shall be provided.	4.1.3.5 Reactor Coolant Pressure Boundary Surveillance Criterion: 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. (GDC 36)	Reactor Coolant Pressure Boundary Surveillance Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leak-tight 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. (GDC 36 of 7/11/67) Inspection of Emergency Core Cooling System (Criterion 36) Criterion: The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system
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. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unob-structed discharge from both ends.	6.1.1.1 Engineered Safety Features Basis for Design Criterion: 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. (GDC 37)	Testing of Emergency Core Cooling System (Criterion 37) Criterion: The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practicable, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

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Criterion 38-Reliability and Testability of Engineered Safety Features (Category A). All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.	6.1.1.2 Reliability and Testability of Engineered Safety Features Criterion: 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. (GDC 38)	2) GDC 38: In order to satisfy the requirements of GDC 38, the calculated pressure at 24 hours should be less than 50% of the peak calculated value. (This is related to the criteria for doses at 24 hours.) *NOTE: Criterion from 10 CFR 50, Appendix A, 1971. Containment Heat Removal (Criterion 38) Criterion: A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptable low levels. 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 not available) the system safety function
Criterion 39-Emergency Power for Engineered Safety Features (Category A). Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.	8.1.1.2 Emergency Power Criterion: 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 component. (GDC 39 and GDC 24)	Inspection of Containment Heat Removal System (Criterion 39) Criterion: The containment heat removal system shall be designed to permit appropriate periodic inspection of important components such as torus, sumps, spray nozzles and piping to assure the integrity and capability of the system.
Criterion 40-Missile Protection (Category A). Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.	4.1.2.4 Missile Protection Criterion: Adequate protection for those engineered safety features, the failures 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. (GDC 40)	Testing of Containment Heat Removal System (Criterion 40) Criterion: The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure: 1) The structural and leaktight integrity of its components. 2) The operability and performance of the active components of the system. 3) The operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.
Criterion 41-Engineered Safety Features Performance Capability (Category A). Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.	6.1.1.4 Engineered Safety Features Performance Capability Criterion: 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. (GDC 41)	Containment Atmosphere Cleanup (Criterion 41) Criterion: Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities 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 not available) its safety function can be accomplished, assuming a single failure. Engineered Safety Features Performance Capability Criterion: 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. (GDC 41 of 7/11/67)
Criterion 42-Engineered Safety Features Components Capability (Category A). Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.	6.1.1.5 Engineered Safety Features Components Capability Criterion: 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. (GDC 42)	Engineered Safety Features Components Capability Criterion: 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 Inspection of Containment Atmosphere Cleanup Systems (Criterion 42) Criterion: The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

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Criterion 43-Accident Aggravation Prevention (Category A). Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after effects of the loss of normal cooling is avoided.	6.1.1.6 Accident Aggravation Prevention Criterion: 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. (GDC 43)	Testing of Containment Atmosphere Cleanup Systems (Criterion 43) Criterion: The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems. Accident Aggravation Prevention Criterion: 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. (GDC 43 of 7/11/67)
Criterion 44-Emergency Core Cooling Systems Capability (Category A). At least two emergency core Cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emer-gency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emer-gency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core Cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or com-ponent to perform its required function can be readily ascertained during reactor opera-tion, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function Is not impaired by the effects of a loss-of-coolant accident and to not lost dur-ing the entire period this function Is required following the accident.	6.2.1.1 Emergency Core Cooling System Capability Criterion: 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. (GDC 44)	Cooling Water (Criterion 44) Criterion: A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation 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 not available) the system safety function can be accomplished, assuming a single failure Criterion 44: 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.
Criterion 45-Inspection of Emergency Core Cooling Systems (Category A).Design provisions shall be made to facilitate physical inspection of all critical parts of the emer-gency core cooling systems, including reactor vessel internals and water infection nozzles.	6.2.1.2 Inspection of Emergency Core Cooling System Criterion: 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. (GDC 45)	Inspection of Emergency Core Cooling System Criterion 45: Design provisions shall, where practical, be made to facilitate inspection of all physical parts of the Emergency Core Cooling System, including reactor vessel internals and water injection nozzles
Criterion 46-Testing of Emergency Core Cooling Systems Components (Category A). Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.	6.2.1.3 Testing of Emergency Core Cooling System Component Criterion: Design provisions shall be made so that components of the emergency core cooling system can be tested periodically for operability and functional performance. (GDC 46)	Testing of Cooling Water System (Criterion 46) Criterion: The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.
Criterion 47-Testing of Emergency Core Cooling Systems (Category A). A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.	6.2.1.4 Testing of Emergency Core Cooling System Criterion: 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. (GDC 47)	Testing of Emergency Core Cooling System Criterion 47: 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.
Criterion 48-Testing of Operational Sequence of Emergency Core Cooling Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.	6.2.1.5 Testing of Operational Sequence of Emergency Core Cooling System Criterion: 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.	Testing of Operational Sequence of Emergency Core Cooling System Criterion 48: 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.
Criterion 49-Containment Design Basis (Category A).The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate with-out exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency Core cooling systems.	5.1.1.1.6 Reactor Containment Design Basis Criterion: The reactor containment structure, including 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. (GDC 49)	Reactor Containment Design Basis Criterion: The reactor containment structure, including 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. (GDC 49 of 7/11/67)

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Comparision of Published (32 FR 10213) General Design Criteria
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Criterion 50-NDT Requirement for Containment Material (Category A) Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30 degrees F above nil ductility transition (NDT) temperature.	**5.1.1.1.7 Nil-ductility Transition Temperature Requirement for Containment Material** **Criterion:** The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50).	Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50 of 7/11/67) 1.3.5 Reactor Containment (Criteria 50 to 57) Containment Design Basis (Criterion 50) Criterion: The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by 50.44, energy from metal, water and other chemical reactions that may result from degradation, but not total failure, of emergency core cooling functioning; (2) the limited experience and experimental data available for defining accident phenomena and containment responses; and (3) the conservatism of the calculational model and input parameters.
Criterion 51-Reactor Coolant Pressure Boundary Outside Containment (Category A). If part of the reactor coolant pressure, boundary is outside the containment, appro-priate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of fea-tures such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.	Compliance not addressed	Fracture Prevention of Containment Pressure Boundary (Criterion 51) Criterion: The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual steady-state, and transient stresses, and (3) size of
Criterion 52-Containment Heat Removal Systems (Category A). Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.	**6.3.1.1 Containment Heat Removal Systems** **Criterion: 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. (GDC 52)**	Criterion: 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. (GDC 52 of 7/11/67) Capability for Containment Leakage Rate Testing (Criterion 52) Criterion: The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.
Criterion 53-Containment Isolation Valves (Category A). Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.	Compliance not addressed	Provisions for Containment Testing and Inspection (Criterion 53) Criterion: The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.
Criterion 54-Containment Leakage Rate Testing (Category A) Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.	5.1.9.1 Initial Containment Leakage Rate Testing Criterion: 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. (GDC 54)	Initial Containment Leakage Rate Testing Criterion: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result form 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. (GDC 54 of 7/11/67) Piping Systems Penetrating Containment (Criterion 54) Criterion: Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

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Criterion 55-Containment Periodic Leakage Rate Testing (Category A). The Containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.	5.1.9.2 Periodic Containment Leakage Rate Testing Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55)	Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55) Criterion: Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: 1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or 2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or 3) One locked closed isolation valve inside and one automatic isolation valve outside of containment. A simple check valve may not be used as the automatic isolation valve outside containment; or 4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and be site environs. Periodic Containment Leakage Rate Testing Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55 of 7/11/67)
Criterion 56 Provisions for Testing of Penetrations (Category A). Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.	5.1.9.3 Provisions for Testing of Penetrations Criterion: 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. (GDC 56)	Provisions for Testing of Penetrations Criterion: 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. (GDC 56 of 7/11/67) Primary Containment Isolation (Criterion 56) Criterion: Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provision for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: 1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or 2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or 3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or 4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
Criterion 57-Provisions for Testing of Isolation Valves (Category A). Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for estab-lishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.	5.1.9.4 Provisions for Testing of Isolation Valves Criterion: 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. (GDC 57)	Closed System Isolation Valves (Criterion 57) Criterion: Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve. Provisions for Testing of Isolation Valves Criterion: 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. (GDC 57 of 7/11/67)
Criterion 58 Inspection of Containment Pressure-Reducing Systems (Category A). Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.	6.4.1.2 Inspection of Containment Pressure-Reducing Systems Criterion: Design provisions shall be made to extent practical to facilitate the periodic physical inspection of all important components of the containment pressure reducing systems, such as pumps, valves, spray nozzles, torus, and sumps. (GDC 58)	Criterion: Design provisions shall be made to the extent practical to facilitate the periodic physical inspection of all important components of the containment pressure reducing systems, such as pumps, valves, spray nozzles and sumps. (GDC 58 of 7/11/67
Criterion 59-Testing of Containment Pressure-Reducing Systems Components (Category A). The containment pressure reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.	6.3.1.3 Testing of the Containment Pressure-Reducing Systems Components Criterion: The containment pressure-reducing systems shall be designed, to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)	Testing of Containment Pressure Reducing Systems Components Criterion: The containment pressure reducing systems shall be designed, to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59 of 7/11/67)

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Criterion 60-Testing of Containment Spray Systems (Category A). A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.	6.3.1.4 Testing of Containment Spray Systems Criterion: 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. (GDC 60)	Control of Releases of Radioactive Materials to the Environment (Criterion 60) Criterion: 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 61-Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A). A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.	6.3.1.5 Testing of Operational Sequence of Containment Pressure-Reducing Systems Criterion: 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 the transfer to alternate power sources. (GDC 61)	Testing of Operational Sequence of Containment Pressure-Reducing Systems Criterion: 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 the transfer to alternate power sources. (GDC 61 of 7/11/67) Fuel Storage and Handling and Radioactivity Control (Criterion 61) Criterion: The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety; (2) with suitable shielding for radiation protection; (3) with appropriate containment, confinement, and filtering systems; (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.
Criterion 62-Inspection of Air Cleanup Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.	6.4.1.5 Inspection of Air Cleanup Systems Criterion: Design provisions shall be made to the extent practical to facilitate physical inspection of all critical parts of containment air cleanup system, such as, ducts, filters, fans, and dampers. (GDC 62)	Prevention of Criticality in Fuel Storage and Handling (Criterion 62) Criterion: Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.
Criterion 63-Testing of Air Cleanup Systems Components (Category A). Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers; can be tested periodically for operability and required functional performance.	6.4.1.6 Testing of Air Cleanup Systems Components Criterion: Design provisions shall be made to the extent practical so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performances. (GDC 63)	Monitoring Fuel and Waste Storage (Criterion 63) Criterion: Appropriate systems shall be provided in the fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in the loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.
Criterion 64-Testing of Air Cleanup Systems (Category A). A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping mate-rials have not deteriorated beyond acceptable limits.	Compliance not addressed	Monitoring Radioactivity Releases (Criterion 64) Criterion: Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA 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 Testing Air Cleanup Systems Criterion: A capability shall be provided to the extent practical for insite periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits. (GDC 64 of 7/11/67)
Criterion 65-Testing of Operational Sequence of Air Cleanup Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.	1.3.7 Engineered Safety Features (GDC 37 - GDC 65) The design, fabrication, testing and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends as discussed in Section 14.3.3.3. They are also designed to cope with any steam or feedwater line break up to and including the main steam or feedwater headers as discussed in Section 14.2.5. The total loss of all offsite power is assumed concurrent with these accidents.	Testing of Operational Sequence of Air Cleanup Systems Criterion: A capability shall be provided to test initially under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability. (GDC 65 of 7/11/67)
Criterion 66-Prevention of Fuel Storage Criticality (Category B). Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.	9.5.1.1 Prevention of Fuel Storage Criticality Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. (GDC 66)	Prevention of Fuel Storage Criticality Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls (GDC 66 of 7/11/67)

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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 that could result In radioactivity release to plant operating areas or the public environs.	9.5.1.2 Fuel and Waste Storage Decay Heat Criterion: 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. (GDC 67)	Criterion: 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 to the public. (GDC 67 of 7/11/67)
Criterion 68-Fuel and Waste Storage Radiation Shielding (Category B). Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.	11.2.1.3 Fuel and Waste Storage Radiation Shielding Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)	Fuel and Waste Storage Radiation Shielding Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities (GDC 68 of 6/11/67).
Criterion 69-Protection Against Radioactivity Release From Spent Fuel and Waste Storage (Category B). Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.	9.5.1.4 Protection Against Radioactivity Release from Spent Fuel and Waste Storage Criterion: 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. (GDC 69)	Protection against Radioactivity Release from Spent Fuel and Waste Storage Criterion: Provisions shall be made in the design of the fuel and waste storage facilities such that no undue risk to the health and safety of the public could results from an accidental release of radioactivity (GDC 69 of 7/11/67)
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 poten-tial reactor accidents of exceedingly low probability of occurrence 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.	Control of Releases of Radioactivity to the Environment Criterion: 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 must 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 dose level guidelines for potential reactor accidents of exceedingly low probability of occurrence (GDC 70).	Control of Releases of Radioactivity to the Environment Criterion: 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 must 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. (GDC 70 of 7/11/67)
(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201) Dated at Washington, D.C., this 28th day of June 1967. For the Atomic Energy Commission. W. B. MCCOOL, Secretary.		

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Paul M. Blanch

Overview

A 35+ year professional presently consulting to the top management of Northeast Utilities Millstone Nuclear Power Station, Indian Point and Maine Yankee and with a distinguished career as an engineer, engineering manager and project coordinator for the construction of nuclear power plants.

Experience

EMPLOYEE CONCERNS AND SAFETY CONCIOUS WORK ENVIRONMENT CONSULTANT - - February 2001 to Present

Consultant reporting to the Chief Nuclear Officer at Indian Point Unit 2 assisting in the evaluation of the plant's Employee Concerns Program and an assessment of the Safety Conscious Work Environment. (SCWE) Work also includes assisting investigations of allegations related to employee discrimination and other technical and safety issues. Developed and implemented training programs for ECP and other site personnel.

EMPLOYEE CONCERNS AND SAFETY CONCIOUS WORK ENVIRONMENT CONSULTANT - - September 2001 to Present

Consultant reporting to the President of Maine Yankee Atomic Power Company. Primary responsibilities include the re-establishment of a Safety Conscious Work Environment (SCWE) and to act as an independent facilitator to resolve differences between employees and management. Evaluated the Employee Concerns Program making recommendations for improvement to the President. Conducted independent investigations of allegations received internally and referral allegations from the NRC.

EMPLOYEE CONCERNS AND SAFETY CONCIOUS WORK ENVIRONMENT CONSULTANT - - February 1997 to 2001

Consultant reporting to the President of Northeast Nuclear Energy Company assisting in the recovery of the three Millstone Units previously on the NRC's "Watch List." Primary responsibilities include the re-establishment of a Safety Conscious Work Environment (SCWE) and to act as an independent facilitator to resolve differences between employees and management. Coordinate many different groups at Millstone including executive management, legal, human resources and the Employee Concerns organization.

Resolve differences at the lowest possible management level. Coordinate with ECP to investigate safety, technical and HIRD issues and review outcomes to assure the investigation was conducted in an unbiased, fair and equitable manner. Coordinate corrective action with the appropriate management, legal and technical organizations.

Work closely with top management and corporate communications to coordinate efforts to regain public confidence with the operation and management of the Millstone site. Provide assistance with regulatory compliance issues and interface with various public interest groups in the Millstone area including State oversight and groups critical of the Millstone operations. Provide both formal and informal feedback to the NRC about the recovery of Millstone and the establishment of a Safety Conscious Work Environment.

Conduct training and make presentations to top nuclear executives about the need to maintain a Safety Conscious Work Environment when requested by the Nuclear Energy Institute and the Nuclear Regulatory Commission.

Made regular presentations to public interest groups, State of Connecticut oversight organizations and the Nuclear Regulatory Commission as to my personal assessment of the work environment at Millstone and the status of corrective actions.

Worked as a team member with other Millstone management providing overall strategic direction to the President to assist in the recovery of Millstone with specific emphasis on public confidence and the establishment of a SCWE.

Provide routine advice to outside legal organizations and other nuclear utility management with respect to dealing with employees raising safety concerns.

Conducted presentations (September 1999 and September 2000) to the Employee Concerns Program Forum providing a perspective on "whistleblower" issues and what management needs to do to properly address these issues.

Conducted presentation in September 2000, along with NRC Chairman Meserve, to the NRC and the NRC's Inspector General's staff on a proposal to resolve "High profile whistleblower" situations. I am continuing to work with the Nuclear Energy Institute to further refine this concept.

Worked closely with the US General Accounting Office conducting its study related to the NRC's handling of whistleblower issues in the nuclear industry,

ENERGY CONSULTANT -- 1993 to 1997

Provided expert witness testimony and worked with the NRC to change Federal Regulations for the protection of individuals identifying safety issues at nuclear licensed facilities.

Worked with the Office of the Inspector General of the NRC to provide major input to a revision of the recently passed federal "Energy Bill" providing additional protection to Nuclear Whistleblowers. This has been referred to as "the Blanch Amendment" by some personnel within the NRC.

Provided advice to both attorneys and their clients to gain an understanding of the NRC and Department of Labor regulations governing the protection of whistleblowers under the Energy Reorganization Act

NORTHEAST UTILITIES -- 1972 to 1993

Supervisor of Electrical Engineering (Instrument and Control Engineering Branch)
Responsible for programs to assure plant reliability and compliance with NRC regulations. Conducted periodic training of employees and contractors to maintain continued cognizance of all corporate and station procedures and regulations. Worked as both a supervisor of an engineering organization and directed the efforts of Stone and Webster and Bechtel to assure safety and compliance during the design and construction of Millstone Units 2 & 3. Primary interface between NU, Westinghouse and Stone and Webster for the conceptual design of electrical and process instrumentation systems during construction of Millstone Unit 3. Assured compliance with all NRC electrical standards and design criteria. Member of the Millstone Nuclear Review Board responsible to the president to assure compliance with all applicable regulations.

Accomplishments

Directed the development of the first real time instrumentation monitoring system for practical use in commercial nuclear plants to asses the overall safety status of the plant and to provide information to

remote facilities during emergency events. This effort resulted in the identification of many instrumentation problems not previously recognized or considered "undetectable failures." As a result of my efforts, and in face of strong opposition from the vendors and the industry, the NRC issued a Bulletin (90-01) requiring all utilities to monitor Rosemount transmitters used in safety applications. A supplement to the Bulletin was issued at the end of 1992.

Recognized the inability of condensate pots to function under de-pressurization events as a direct result of NU's computerized instrument monitoring system. This is one of the most significant safety issues identified in the nuclear industry. Developed a water injection system into the reference legs that precluded the absorption of these gases. This solution was adopted by the entire nuclear industry.

Developed a program to reduce or eliminate the need for periodic calibration of analog instrumentation and the elimination of the need for pressure transmitter response time testing. The formation of an ISA Standard activity (ISA 67.06) for the development of a standard for Performance Monitoring of Safety Related Instruments in Nuclear Power Plants was a direct result of these efforts.

Received a "First Use" award from Electric Power Research Institute (EPRI) for the application of Signal Validation for the identification of failed sensors during accident, as a direct result of developing and implementing signal validation for emergency computer systems.

Nuclear Operations Engineer (1979 – 1981)

Provide coordination between the Millstone plants and headquarters engineering, design and regulatory affairs department. Appointed as NU's representative to coordinate the NRC's backfit requirements following the TMI accident. Many of these new requirements involved the addition of effluent and area radiation monitors.

Senior I & C Engineer (1974 – 1979)

Specified and directed the design, procurement and installation of instrumentation systems for use in Millstone Units #2 and #3. This included all process instrumentation along with all effluent monitoring systems including process and area radiation monitoring systems.

UNITED STATES NAVY – 1963 to 1971

Electrical plant and Reactor operator and Leading Petty Officer aboard the Nuclear Powered Submarine USS Patrick Henry (SSBN-599). Qualified electrical plant and reactor operator and instructor at Navy prototype reactor (S1C). US Navy Submarine School 1968. US Navy Nuclear Power School 1965. US Navy Electronics Technician School 1964.

Special Qualifications

Actively participated and contributed to two recent studies conducted by the NRC and NU addressing the cultural problems at Northeast Utilities. Collaborated with the Fundamental Cause Assessment Team and the NRC's Millstone Independent Review Group and provided insights as to the root causes of the problems effecting the NU nuclear organization.

Named Utility Engineer of the Year (1993) by Westinghouse Electric and Control Magazine for advancing the safety of nuclear power.

Publicly recognized in October 1992 by the Chairman of the NRC (Ivan Selin) for significant contributions to nuclear safety, related to the identification of the condensate pot problems on Boiling and Pressurized Water Reactors.

Testified before the US Senate Subcommittee about the failure of the NRC's regulatory practices and the NRC's mistreatment of Nuclear Whistleblowers. Instrumental in developing Connecticut's Nuclear Whistleblower Law effective October 1, 1992 which is the strongest Whistleblower Protection Law in the country. Discussed in Time Magazine (March 4, 1996) as a contributor to nuclear safety.

Registered Professional Controls Engineer.

Education

BS Electrical Engineering, Magna Cum Laude, 1972, University of Hartford
Graduate courses in Mechanical and Thermodynamic Engineering
US Navy Submarine School, 1968
US Navy Nuclear Power School, 1965
US Navy Electronics Technician School, 1964

Professional Associations

Member of the ANS Standards Committee (ANS 6.8.1 and 6.8.2) responsible for developing and specifying the requirements for process, effluent and area radiation monitors for commercial nuclear power plants.

Vice Chairman, Institute of Nuclear Power Operations (INPO) Two Standards Activities in response to Three Mile Island including Post Accident Monitoring requirements.

Member of the ANS Standards Committee responsible for developing the requirements for seismic monitoring systems for nuclear power plants.

Chairman of Two Committees for the Institute for Nuclear Power Operations (INPO) related to Three Mile Island post accident monitoring requirements and emergency response facilities.

Member of ISA 67.04 for the development of Instrument Setpoints for Nuclear Power Plants

Registered Professional Engineer - California

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Operations, Inc.

Docket Nos. 50-247-LR, 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

DECLARATION OF DAVID A. SCHLISSEL

David A. Schlissel, hereby declares under penalty of perjury that the following is true and correct:

1. I am a senior consultant at Synapse Energy Economics, Inc. (Synapse), an energy and economic consulting firm located in Cambridge, Massachusetts.

2. Synapse has been retained by the New York State Office of the Attorney General to provide expert services to the State of New York concerning the proposed relicensing of the two operating reactors located at the Indian Point Nuclear Power Station in the Village of Buchanan in Westchester County (Indian Point Unit 2 and Indian Point Unit 3).

3. When it is in service, Indian Point Unit 2 can produce up to 1,028 MW per year; Indian Point Unit 3 can produce up to 1,041 MW when it is in service. The Indian Point nuclear reactors, however, cannot run indefinitely. Approximately every 24 months, each reactor is taken off line for refueling and maintenance work. According to the Entergy's recent investor report, over the last two years, planned outages for maintenance and refueling at Indian Point Unit 2 and Unit 3 have lasted approximately three to four weeks (24 to 31 days). *See* Entergy Statistical Report and Investor Guide 2006, p. 52. In addition, from time to time, each unit may experience unplanned outages.

4. Attached hereto and made a part of this sworn statement is a report prepared by me concerning readily-available means to replace the power generated by Indian Point Unit 2 and/or Indian Point Unit 3. This report examines the availability of: (1) energy conservation and efficiency measures; (2) repowering of existing power plants; (3) renewable energy resources; (4) certain transmission system upgrades and enhancements; and (5) the potential for the addition of new generating facilities. See Synapse Energy Economics, Inc, " Report on the Availability of Replacement Capacity and Energy for Indian Point Units 2 and 3" (November 28, 2007).

5. To prepare the attached report, my staff and I have examined various publicly-available information, including, but not limited to, reports prepared by the

New York State Energy Research and Development Authority, the New York Independent System Operator, the New York State Department of Public Service, the U.S. Department of Energy, the U.S. Nuclear Regulatory Commission, Levitan & Associates for the County of Westchester, the New York State Reliability Council, and the National Academy of Sciences. I also examined the April 30, 2007 License Renewal Application filed by Entergy, the accompanying Environmental Report, and the Entergy Statistical Report and Investor Guide 2006.

6. The report that I prepared concludes that the capacity and energy provided by Indian Point Units 2 and 3 can be replaced if the Units are not relicensed. In particular, energy efficiency, renewable resources, the repowering of older generating facilities, transmission upgrades and new natural gas-fired generating facilities represent viable alternatives to the relicensing of Indian Point. Substantial reductions in peak demand and energy requirements will be achieved by 2013 under the state's newly announced "15 by 15" Clean Energy Plan. Significant amounts of new renewable resources will be available as a result of the state's renewable energy portfolio standard and other initiatives. In addition, thousands of megawatts ("MW") of new generating capacity can be provided by the repowering (i.e., rebuilding) of older generating facilities both along the Hudson River and in the downstate area of the state in New York City and on Long Island. At the same time, transmission system upgrades also can increase the amounts of

power that can provided to the downstate region of the State. Finally, there is the potential for the addition of several thousand megawatts of new generating capacity in the Hudson River Valley and in downstate New York.

7. Also attached hereto is a copy of my current Curriculum Vitae (CV).

8. The report and CV are true and correct to the best of my personal knowledge.

9. Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated:

November 28, 2007
Cambridge, Massachusetts

David A. Schlissel

David A. Schlissel

David A. Schlissel

Senior Consultant
Synapse Energy Economics
22 Pearl Street, Cambridge, MA 02139
(617) 661-3248 ext. 224 • Fax: (617) 661-0599
www.synapse-energy.com
dschlissel@synapse-energy.com

SUMMARY

I have worked for thirty years as a consultant and attorney on complex management, engineering, and economic issues, primarily in the field of energy. This work has involved conducting technical investigations, preparing economic analyses, presenting expert testimony, providing support during all phases of regulatory proceedings and litigation, and advising clients during settlement negotiations. I received undergraduate and advanced engineering degrees from the Massachusetts Institute of Technology and Stanford University, respectively, and a law degree from Stanford Law School

PROFESSIONAL EXPERIENCE

Electric System Reliability - Evaluated whether new transmission lines and generation facilities were needed to ensure adequate levels of system reliability. Investigated the causes of distribution system outages and inadequate service reliability. Examined the reasonableness of utility system reliability expenditures.

Transmission Line Siting – Examined the need for proposed transmission lines. Analyzed whether proposed transmission lines could be installed underground. Worked with clients to develop alternate routings for proposed lines that would have reduced impacts on the environment and communities.

Power Plant Operations and Economics - Investigated the causes of more than one hundred power plant and system outages, equipment failures, and component degradation, determined whether these problems could have been anticipated and avoided, and assessed liability for repair and replacement costs. Examined power plant operating, maintenance, and capital costs. Analyzed power plant operating data from the NERC Generating Availability Data System (GADS). Evaluated utility plans for and management of the replacement of major power plant components. Assessed the adequacy of power plant quality assurance and maintenance programs. Examined the selection and supervision of contractors and subcontractors.

Power Plant Repowering - Evaluated the environmental, economic and reliability impacts of rebuilding older, inefficient generating facilities with new combined cycle technology.

Power Plant Air Emissions – Investigated whether proposed generating facilities would provide environmental benefits in terms of reduced emissions of NO_x, SO_2 and CO_2. Examined whether new state emission standards would lead to the retirement of existing power plants or otherwise have an adverse impact on electric system reliability.

Power Plant Water Use – Examined power plant repowering as a strategy for reducing water consumption at existing electric generating facilities. Analyzed the impact of converting power plants from once-through to closed-loop systems with cooling towers on plant revenues and electric system reliability. Evaluated the potential impact of the EPA's Proosed Clean Water Act Section 316(b) Rule for Cooling Water Intake Structures at existing power plants.

Nuclear Power - Examined the impact of the nuclear power plant life extensions and power uprates on decommissioning costs and collections policies. Evaluated utility decommissioning cost estimates and cost collection plans. Examined the reasonableness of utility decisions to sell nuclear power assets and evaluated the value received as a result of the auctioning of those plants. Investigated the significance of the increasing ownership of nuclear power plants by multiple tiered holding companies with limited liability company subsidiaries. Investigated the potential safety consequences of nuclear power plant structure, system, and component failures.

Electric Industry Regulation and Markets - Investigated whether new generating facilities that were built for a deregulated subsidiary should be included in the rate base of a regulated utility. Evaluated the reasonableness of proposed utility power purchase agreements with deregulated affiliates. Investigated the prudence of utility power purchases in deregulated markets. Examined whether generating facilities experienced more outages following the transition to a deregulated wholesale market in New England. Evaluated the reasonableness of nuclear and fossil plant sales, auctions, and power purchase agreements. Analyzed the impact of proposed utility mergers on market power. Assessed the reasonableness of contract provisions and terms in proposed power supply agreements.

Economic Analysis - Analyzed the costs and benefits of energy supply options. Examined the economic and system reliability consequences of the early retirement of major electric generating facilities. Evaluated whether new electric generating facilities are used and useful. Quantified replacement power costs and the increased capital and operating costs due to identified instances of mismanagement.

Expert Testimony - Presented the results of management, technical and economic analyses as testimony in more than ninety proceedings before regulatory boards and commissions in twenty three states, before two federal regulatory agencies, and in state and federal court proceedings.

Litigation and Regulatory Support - Participated in all aspects of the development and preparation of case presentations on complex management, technical, and economic issues. Assisted in the preparation and conduct of pre-trial discovery and depositions. Helped identify and prepare expert witnesses. Aided the preparation of pre-hearing petitions and motions and post-hearing briefs and appeals. Assisted counsel in preparing for hearings and oral arguments. Advised counsel during settlement negotiations.

TESTIMONY, AFFIDAVITS, DEPOSITIONS AND COMMENTS

West Virginia Public Service Commission (Case No. 06-0033-E-CN) – November 2007
Appalachian Power Company's application for a Certificate of Public Convenience and Necessity for a 600 MW integrated gasification combined cycle generating facility.

Iowa Utility Board (Docket No. GCU-07-01) – October 2007
Whether Interstate Power & Light Company's adequately considered the risks associated with building a new coal-fired power plant and whether that Company's participation in the proposed Marshalltown plant is prudent.

Virginia State Corporation Commission (Case No. PUE-2007-00066) – November 2007
Whether Dominion Virginia Power's adequately considered the risks associated with building the proposed Wise County coal-fired power plant and whether that Commission should grant a certificate of public convenience and necessity for the plant.

Louisiana Public Service Commission (Docket No. U-30192) – September 2007
The reasonableness of Entergy Louisiana's proposal to repower the Little Gypsy Unit 3 generating facility as a coal-fired power plant.

Arkansas Public Service Commission (Docket No. 06-154-U) – July 2007
The probable economic impact of the Southwestern Electric Power Company's proposed Hempstead coal-fired power plant project.

North Dakota Public Service Commission (Case Nos. PU-06-481 and 482) – May 2007
Whether the participation of Otter Tail Power Company and Montana-Dakota Utilities in the Big Stone II Generating Project is prudent.

Indiana Utility Regulatory Commission (Cause No. 43114) – May 2007
The appropriate carbon dioxide ("CO_2") emissions prices that should be used to analyze the relative economic costs and benefits of Duke Energy Indiana and Vectren Energy Delivery of Indiana's proposed Integrated Gasification Combined Cycle Facility and whether Duke and Vectren have appropriately reflected the capital cost of the proposed facility in their modeling analyses.

Public Service Commission of Wisconsin (Docket No. 6630-EI-113) – March 2007
Whether the proposed sale of the Point Beach Nuclear Plant to FPL Energy Point Beach, LLC, is in the interest of the ratepayers of Wisconsin Electric Power Company.

Florida Public Service Commission (Docket No. 070098-EI) – March 2007
Florida Light & Power Company's need for and the economics of the proposed Glades Power Park.

Michigan Public Service Commission (Case No. 14992-U) – December 2006
The reasonableness of the proposed sale of the Palisades Nuclear Power Plant.

Minnesota Public Utilities Commission (Docket No. CN-05-619) – November 2006
Whether the co-owners of the proposed Big Stone II coal-fired generating plant have appropriately reflected the potential for the regulation of greenhouse gases in their analyses of the facility; and whether the proposed project is a lower cost alternative than renewable options, conservation and load management.

North Carolina Utilities Commission (Docket No. E-7, Sub 790) – September 2006 and January 2007
Duke's need for two new 800 MW coal-fired generating units and the relative economics of adding these facilities as compared to other available options including energy efficiency and renewable technologies.

New Mexico Public Regulatory Commission (Case No. 05-00275-UT) – September 2006
Report to the New Mexico Commission on whether the settlement value of the adjustment for moving the 141 MW Afton combustion turbine merchant plant into rate base is reasonable.

Arizona Corporation Commission (Docket No. E-01345A-0816) – August and September 2006
Whether APS's acquisition of the Sundance Generating Station was prudent and the reasonableness of the amounts that APS requested for fossil plant O&M.

U.S. District Court for the District of Montana (Billings Generation, Inc. vs. Electrical Controls, Inc, et al., CV-04-123-BLG-RFC) – August 2006
Quantification of plaintiff's business losses during an extended power plant outage and plaintiff's business earnings due to the shortening and delay of future plant outages. [Confidential Expert Report]

Deposition in South Dakota Public Utility Commission Case No. EL05-022 – June 14, 2006

South Dakota Public Utility Commission (Case No. EL05-022) – May and June 2006
Whether the co-owners of the proposed Big Stone II coal-fired generating plant have appropriately reflected the potential for the regulation of greenhouse gases in their analyses of the alternatives to the proposed facility; the need and timing for new supply options in the co-owners' service territories; and whether there are alternatives to the proposed facility that are technically feasible and economically cost-effective.

Georgia Public Service Commission (Docket No. 22449-U) – May 2006
Georgia Power Company's request for an accounting order to record early site permitting and construction operating license costs for new nuclear power plants.

California Public Utilities Commission (Dockets Nos. A.05-11-008 and A.05-11-009) – April 2006
The estimated costs for decommissioning the Diablo Canyon, SONGS 2&3 and Palo Verde nuclear power plants and the annual contributions that are needed from ratepayers to assure that adequate funds will be available to decommission these plants at the projected ends of their service lives.

New Jersey Board of Public Utilities (Docket No. EM05020106) – November and December 2005 and March 2006
Joint Testimony with Bob Fagan and Bruce Biewald on the market power implications of the proposed merger between Exelon Corp. and Public Service Enterprise Group.

Virginia State Corporation Commission (Case No. PUE-2005-00018)– November 2005
The siting of a proposed 230 kV transmission line.

Iowa Utility Board (Docket No. SPU-05-15) – September and October 2005
The reasonableness of IPL's proposed sale of the Duane Arnold Energy Center nuclear plant.

New York State Department of Environmental Conservation (DEC #3-3346-00011/00002) – October 2005
The likely profits that Dynegy will earn from the sale of the energy and capacity of the Danskammer Generating Facility if the plant is converted from once-through to closed-cycle cooling with wet towers or to dry cooling.

Arkansas Public Service Commission (Docket 05-042-U) – July and August 2005
Arkansas Electric Cooperative Corporation's proposed purchase of the Wrightsville Power Facility.

Maine Public Utilities Commission (Docket No. 2005-17) – July 2005
Joint testimony with Peter Lanzalotta and Bob Fagan evaluating Eastern Maine Electric Cooperative's request for a CPCN to purchase 15 MW of transmission capacity from New Brunswick Power.

Federal Energy Regulatory Commission (Docket No. EC05-43-0000) – April and May 2005
Joint Affidavit and Supplemental Affidavit with Bruce Biewald on the market power aspects of the proposed merger of Exelon Corporation and Public Service Enterprise Group, Inc.

Maine Public Utilities Commission (Docket No. 2004-538 Phase II) – April 2005
Joint testimony with Peter Lanzalotta and Bob Fagan evaluating Maine Public Service Company's request for a CPCN to purchase 35 MW of transmission capacity from New Brunswick Power.

Maine Public Utilities Commission (Docket No. 2004-771) – March 2005
Analysis of Bangor Hydro-Electric's Petition for a Certificate of Public Convenience and Necessity to construct a 345 kV transmission line

United States District Court for the Southern District of Ohio, Eastern Division (Consolidated Civil Actions Nos. C2-99-1182 and C2-99-1250)
Whether the public release of company documents more than three years old would cause competitive harm to the American Electric Power Company. [Confidential Expert Report]

New Jersey Board of Public Utilities (Docket No. EO03121014) – February 2005
Whether the Board of Public Utilities can halt further collections from Jersey Central Power & Light Company's ratepayers because there already are adequate funds in the company's decommissioning trusts for the Three Mile Island Unit No. 2 Nuclear Plant to allow for the decommissioning of that unit without endangered the public health and safety.

Maine Public Utilities Commission (Docket No. 2004-538) – January and March 2005
Analysis of Maine Public Service Company's request to construct a 138 kV transmission line from Limestone, Maine to the Canadian Border.

California Public Utilities Commission (Application No. AO4-02-026) – December 2004 and January 2005
Southern California Edison's proposed replacement of the steam generators at the San Onofre Unit 2 and Unit 3 nuclear power plants and whether the utility was imprudent for failing to initiate litigation against Combustion Engineering due to defects in the design of and materials used in those steam generators.

United States District Court for the Southern District of Indiana, Indianapolis Division (Civil Action No. IP99-1693) – December 2004
Whether the public release of company documents more than three years old would cause competitive harm to the Cinergy Corporation. [Confidential Expert Report]

California Public Utilities Commission (Application No. AO4-01-009) – August 2004
Pacific Gas & Electric's proposed replacement of the steam generators at the Diablo Canyon nuclear power plant and whether the utility was imprudent for failing to initiate litigation against Westinghouse due to defects in the design of and materials used in those steam generators.

Public Service Commission of Wisconsin (Docket No. 6690-CE-187) – June, July and August 2004
Whether Wisconsin Public Service Corporation's request for approval to build a proposed 515 MW coal-burning generating facility should be granted.

Public Service Commission of Wisconsin (Docket No. 05-EI-136) – May and June 2004
Whether the proposed sale of the Kewaunee Nuclear Power Plant to a subsidiary of an out-of-state holding company is in the public interest.

Connecticut Siting Council (Docket No. 272) – May 2004
Whether there are technically viable alternatives to the proposed 345-kV transmission line between Middletown and Norwalk Connecticut and the length of the line that can be installed underground.

Arizona Corporation Commission (Docket No. E-01345A-03-0437 – February 2004
Whether Arizona Public Service Company should be allowed to acquire and include in rate base five generating units that were built by a deregulated affiliate.

State of Rhode Island Energy Facilities Siting Board (Docket No. SB-2003-1) – February 2004
Whether the cost of undergrounding a relocated 115kV transmission line would be eligible for regional cost socialization.

State of Maine Department of Environmental Protection (Docket No. A-82-75-0-X) – December 2003
The storage of irradiated nuclear fuel in an Independent Spent Fuel Storage Installation (ISFSI) and whether such an installation represents an air pollution control facility.

Rhode Island Public Utility Commission (Docket No. 3564) – December 2003 and January 2004
Whether Narragansett Electric Company should be required to install a relocated 115kV transmission line underground.

New York State Board on Electric Generation Siting and the Environment (Case No. 01-F-1276) – September, October and November 2003
The environmental, economic and system reliability benefits that can reasonably be expected from the proposed 1,100 MW TransGas Energy generating facility in Brooklyn, New York.

Wisconsin Public Service Commission (Case 6690-UR-115209) - September and October 2003
The reasonableness of Wisconsin Public Service Corporation's decommissioning cost collections for the Kewaunee Nuclear Plant.

Oklahoma Corporation Commission (Cause No. 2003-121) – July 2003
Whether Empire District Electric Company properly reduced its capital costs to reflect the write-off of a portion of the cost of building a new electric generating facility.

Arkansas Public Service Commission (Docket 02-248-U) – May 2003
Entergy's proposed replacement of the steam generators and the reactor vessel head at the ANO Unit 1 Steam Generating Station.

Appellate Tax Board, State of Massachusetts (Docket No C258405-406) – May 2003
The physical nature of electricity and whether electricity is a tangible product or a service.

Maine Public Utilities Commission (Docket 2002-665-U) – April 2003
Analysis of Central Maine Power Company's proposed transmission line for Southern York County and recommendation of alternatives.

Massachusetts Legislature, Joint Committees on Government Regulations and Energy – March 2003
Whether PG&E can decide to permanently retire one or more of the generating units at its Salem Harbor Station if it is not granted an extension beyond October 2004 to reduce the emissions from the Station's three coal-fired units and one oil-fired unit.

New Jersey Board of Public Utilities (Docket No. ER02080614) – January 2003
The prudence of Rockland Electric Company's power purchases during the period August 1, 1999 through July 31, 2002.

New York State Board on Electric Generation Siting and the Environment (Case No. 00-F-1356) – September and October 2002 and January 2003
The need for and the environmental benefits from the proposed 300 MW Kings Park Energy generating facility.

Arizona Corporation Commission (Docket No. E-01345A-01-0822) – March 2002
The reasonableness of Arizona Public Service Company's proposed long-term power purchase agreement with an affiliated company.

New York State Board on Electric Generation Siting and the Environment (Case No. 99-F-1627) – March 2002
Repowering NYPA's existing Poletti Station in Queens, New York.

Connecticut Siting Council (Docket No. 217) – March 2002, November 2002, and January 2003
Whether the proposed 345-kV transmission line between Plumtree and Norwalk substations in Southwestern Connecticut is needed and will produce public benefits.

Vermont Public Service Board (Case No. 6545) – January 2002
Whether the proposed sale of the Vermont Yankee Nuclear Plant to Entergy is in the public interest of the State of Vermont and Vermont ratepayers.

Connecticut Department of Public Utility Control (Docket 99-09-12RE02) – December 2001

The reasonableness of adjustments that Connecticut Light and Power Company seeks to make to the proceeds that it received from the sale of Millstone Nuclear Power Station.

Connecticut Siting Council (Docket No. 208) – October 2001
Whether the proposed cross-sound cable between Connecticut and Long Island is needed and will produce public benefits for Connecticut consumers.

New Jersey Board of Public Utilities (Docket No. EM01050308) - September 2001
The market power implications of the proposed merger between Conectiv and Pepco.

Illinois Commerce Commission Docket No. 01-0423 – August, September, and October 2001
Commonwealth Edison Company's management of its distribution and transmission systems.

New York State Board on Electric Generation Siting and the Environment (Case No. 99-F-1627) - August and September 2001
The environmental benefits from the proposed 500 MW NYPA Astoria generating facility.

New York State Board on Electric Generation Siting and the Environment (Case No. 99-F-1191) - June 2001
The environmental benefits from the proposed 1,000 MW Astoria Energy generating facility.

New Jersey Board of Public Utilities (Docket No. EM00110870) - May 2001
The market power implications of the proposed merger between FirstEnergy and GPU Energy.

Connecticut Department of Public Utility Control (Docket 99-09-12RE01) - November 2000
The proposed sale of Millstone Nuclear Station to Dominion Nuclear, Inc.

Illinois Commerce Commission (Docket 00-0361) - August 2000
The impact of nuclear power plant life extensions on Commonwealth Edison Company's decommissioning costs and collections from ratepayers.

Vermont Public Service Board (Docket 6300) - April 2000
Whether the proposed sale of the Vermont Yankee nuclear plant to AmerGen Vermont is in the public interest.

Massachusetts Department of Telecommunications and Energy (Docket 99-107, Phase II) - April and June 2000
The causes of the May 18, 1999, main transformer fire at the Pilgrim generating station.

Connecticut Department of Public Utility Control (Docket 00-01-11) - March and April 2000
The impact of the proposed merger between Northeast Utilities and Con Edison, Inc. on the reliability of the electric service being provided to Connecticut ratepayers.

Connecticut Department of Public Utility Control (Docket 99-09-12) - January 2000
The reasonableness of Northeast Utilities plan for auctioning the Millstone Nuclear Station.

Connecticut Department of Public Utility Control (Docket 99-08-01) - November 1999
Generation, Transmission, and Distribution system reliability.

Illinois Commerce Commission (Docket 99-0115) - September 1999
Commonwealth Edison Company's decommissioning cost estimate for the Zion Nuclear Station.

Connecticut Department of Public Utility Control (Docket 99-03-36) - July 1999
Standard offer rates for Connecticut Light & Power Company.

Connecticut Department of Public Utility Control (Docket 99-03-35) - July 1999
Standard offer rates for United Illuminating Company.

Connecticut Department of Public Utility Control (Docket 99-02-05) - April 1999
Connecticut Light & Power Company stranded costs.

Connecticut Department of Public Utility Control (Docket 99-03-04) - April 1999
United Illuminating Company stranded costs.

Maryland Public Service Commission (Docket 8795) - December 1998
Future operating performance of Delmarva Power Company's nuclear units.

Maryland Public Service Commission (Dockets 8794/8804) - December 1998
Baltimore Gas and Electric Company's proposed replacement of the steam generators at the Calvert Cliffs Nuclear Power Plant. Future performance of nuclear units.

Indiana Utility Regulatory Commission (Docket 38702-FAC-40-S1) - November 1998
Whether the ongoing outages of the two units at the D.C. Cook Nuclear Plant were caused or extended by mismanagement.

Arkansas Public Service Commission (Docket 98-065-U) - October 1998
Entergy's proposed replacement of the steam generators at the ANO Unit 2 Steam Generating Station.

Massachusetts Department of Telecommunications and Energy (Docket 97-120) - October 1998
Western Massachusetts Electric Company's Transition Charge. Whether the extended 1996-1998 outages of the three units at the Millstone Nuclear Station were caused or extended by mismanagement.

Connecticut Department of Public Utility Control (Docket 98-01-02) - September 1998
Nuclear plant operations, operating and capital costs, and system reliability improvement costs.

Illinois Commerce Commission (Docket 97-0015) - May 1998
Whether any of the outages of Commonwealth Edison Company's twelve nuclear units during 1996 were caused or extended by mismanagement. Whether equipment problems, personnel performance weaknesses, and program deficiencies could have been avoided or addressed prior to plant outages. Outage-related fuel and replacement power costs.

Public Service Commission of West Virginia (Case 97-1329-E-CN) - March 1998
The need for a proposed 765 kV transmission line from Wyoming, West Virginia, to Cloverdate, Virginia.

Illinois Commerce Commission (Docket 97-0018) - March 1998
Whether any of the outages of the Clinton Power Station during 1996 were caused or extended by mismanagement.

Connecticut Department of Public Utility Control (Docket 97-05-12) - October 1997
The increased costs resulting from the ongoing outages of the three units at the Millstone Nuclear Station.

New Jersey Board of Public Utilities (Docket ER96030257) - August 1996
Replacement power costs during plant outages.

Illinois Commerce Commission (Docket 95-0119) - February 1996
Whether any of the outages of Commonwealth Edison Company's twelve nuclear units during 1994 were caused or extended by mismanagement. Whether equipment problems, personnel performance weaknesses, and program deficiencies could have been avoided or addressed prior to plant outages. Outage-related fuel and replacement power costs.

Public Utility Commission of Texas (Docket 13170) - December 1994
Whether any of the outages of the River Bend Nuclear Station during the period October 1, 1991, through December 31, 1993, were caused or extended by mismanagement.

Public Utility Commission of Texas (Docket 12820) - October 1994
Operations and maintenance expenses during outages of the South Texas Nuclear Generating Station.

Wisconsin Public Service Commission (Cases 6630-CE-197 and 6630-CE-209) - September and October 1994
The reasonableness of the projected cost and schedule for the replacement of the steam generators at the Point Beach Nuclear Power Plant. The potential impact of plant aging on future operating costs and performance.

Public Utility Commission of Texas (Docket 12700) - June 1994
Whether El Paso Electric Company's share of Palo Verde Unit 3 was needed to ensure adequate levels of system reliability. Whether the Company's investment in Unit 3 could be expected to generate cost savings for ratepayers within a reasonable number of years.

Arizona Corporation Commission (Docket U-1551-93-272) - May and June 1994
Southwest Gas Corporation's plastic and steel pipe repair and replacement programs.

Connecticut Department of Public Utility Control (Docket 92-04-15) - March 1994
Northeast Utilities management of the 1992/1993 replacement of the steam generators at Millstone Unit 2.

Connecticut Department of Public Utility Control (Docket 92-10-03) - August 1993
Whether the 1991 outage of Millstone Unit 3 as a result of the corrosion of safety-related plant piping systems was due to mismanagement.

Public Utility Commission of Texas (Docket 11735) - April and July 1993
Whether any of the outages of the Comanche Peak Unit 1 Nuclear Station during the period August 13, 1990, through June 30, 1992, were caused or extended by mismanagement.

Connecticut Department of Public Utility Control (Docket 91-12-07) - January 1993 and August 1995
Whether the November 6, 1991, pipe rupture at Millstone Unit 2 and the related outages of the Connecticut Yankee and Millstone units were caused or extended by mismanagement. The impact of environmental requirements on power plant design and operation.

Connecticut Department of Public Utility Control (Docket 92-06-05) - September 1992
United Illuminating Company off-system capacity sales. [Confidential Testimony]

Public Utility Commission of Texas (Docket 10894) - August 1992
Whether any of the outages of the River Bend Nuclear Station during the period October 1, 1988, through September 30, 1991, were caused or extended by mismanagement.

Connecticut Department of Public Utility Control (Docket 92-01-05) - August 1992
Whether the July 1991 outage of Millstone Unit 3 due to the fouling of important plant systems by blue mussels was the result of mismanagement.

California Public Utilities Commission (Docket 90-12-018) - November 1991, April 1992, June and July 1993
Whether any of the outages of the three units at the Palo Verde Nuclear Generating Station during 1989 and 1990 were caused or extended by mismanagement. Whether equipment problems, personnel performance weaknesses and program deficiencies could have been avoided or addressed prior to outages. Whether specific plant operating cost and capital expenditures were necessary and prudent.

Public Utility Commission of Texas (Docket 9945) - June 1991
Whether El Paso Electric Company's share of Palo Verde Unit 3 was needed to ensure adequate levels of system reliability. Whether the Company's investment in the unit could be expected to generate cost savings for ratepayers within a reasonable number of years. El Paso Electric Company's management of the planning and licensing of the Arizona Interconnection Project transmission line.

Arizona Corporation Commission (Docket U-1345-90-007) - December 1990 and April 1991
Arizona Public Service Company's management of the planning, construction and operation of the Palo Verde Nuclear Generating Station. The costs resulting from identified instances of mismanagement.

New Jersey Board of Public Utilities (Docket ER89110912J) - July and October 1990
The economic costs and benefits of the early retirement of the Oyster Creek Nuclear Plant. The potential impact of the unit's early retirement on system reliability. The cost and schedule for siting and constructing a replacement natural gas-fired generating plant.

Public Utility Commission of Texas (Docket 9300) - June and July 1990
Texas Utilities management of the design and construction of the Comanche Peak Nuclear Plant. Whether the Company was prudent in repurchasing minority owners' shares of Comanche Peak without examining the costs and benefits of the repurchase for its ratepayers.

Federal Energy Regulatory Commission (Docket EL-88-5-000) - November 1989
Boston Edison's corporate management of the Pilgrim Nuclear Station.

Connecticut Department of Public Utility Control (Docket 89-08-11) - November 1989
United Illuminating Company's off-system capacity sales.

Kansas State Corporation Commission (Case 164,211-U) - April 1989
Whether any of the 127 days of outages of the Wolf Creek generating plant during 1987 and 1988 were the result of mismanagement.

Public Utility Commission of Texas (Docket 8425) - March 1989
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Carolina Power & Light Company's management of the design and construction of the Harris Nuclear Project. The Company's management of quality assurance and quality control activities. The impact of safety-related and environmental requirements on construction costs and schedule. The cost and schedule consequences of identified instances of mismanagement.

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The planned startup and power ascension testing program for the Nine Mile Point Unit 2 generating facility.

Illinois Commerce Commission (Dockets 86-0043 and 86-0096) - April 1987
The reasonableness of certain terms in a proposed Power Supply Agreement.

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The in-service criteria to be used to determine when a new generating facility was capable of providing safe, adequate, reliable and efficient service.

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Northern Indiana Public Service Company's planning for the Schaefer Unit 18 generating facility. Whether the capacity from Unit 18 was needed to ensure adequate system reliability. The rate consequences of excess capacity on the Company's system.

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The radiation effects of low power testing on the structures, equipment and components in a new nuclear power plant.

New York State Public Service Commission (Case 28124) - April 1986 and May 1987
The terms and provisions in a utility's contract with an equipment supplier. The prudence of the utility's planning for a new generating facility. Expenditures on a canceled generating facility.

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The construction schedule for Palo Verde Unit No. 1. Regulatory and technical factors that would likely affect future plant operating costs.

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A performance standard for the Shoreham nuclear power plant.

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The impact of safety-related regulatory requirements and plant aging on power plant operating costs and performance. Regulatory factors and plant-specific design features that will likely affect the future operating costs and performance of the Wolf Creek Nuclear Plant.

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The Potential Safety Consequences of Steam Generator Tube Cracking at the Byron and Braidwood Nuclear Stations, a report for the Environmental Law and Policy Center of the Midwest, 1995.

Report to the Public Policy Group Concerning Future Trojan Nuclear Plant Operating Performance and Costs, July 15, 1992.

Report to the New York State Consumer Protection Board on the Costs of the 1991 Refueling Outage of Indian Point 2, December 1991.

Preliminary Report on Excess Capacity Issues to the Public Utility Regulation Board of the City of El Paso, Texas, April 1991.

Nuclear Power Plant Construction Costs, presentation at the November, 1987, Conference of the National Association of State Utility Consumer Advocates.

Comments on the Final Report of the National Electric Reliability Study, a report for the New York State Consumer Protection Board, February 27, 1981.

OTHER SIGNIFICANT INVESTIGATIONS AND LITIGATION SUPPORT WORK

Reviewed the salt deposition mitigation strategy proposed for Reliant Energy's repowering of its Astoria Generating Station. October 2002 through February 2003.

Assisted the Connecticut Office of Consumer Counsel in reviewing the auction of Connecticut Light & Power Company's power purchase agreements. August and September, 2000.

Assisted the New Jersey Division of the Ratepayer Advocate in evaluating the reasonableness of Atlantic City Electric Company's proposed sale of its fossil generating facilities. June and July, 2000.

Investigated whether the 1996-1998 outages of the three Millstone Nuclear Units were caused or extended by mismanagement. 1997 and 1998. Clients were the Connecticut Office of Consumer Counsel and the Office of the Attorney General of the Commonwealth of Massachusetts.

Investigated whether the 1995-1997 outages of the two units at the Salem Nuclear Station were caused or extended by mismanagement. 1996-1997. Client was the New Jersey Division of the Ratepayer Advocate.

Assisted the Associated Industries of Massachusetts in quantifying the stranded costs associated with utility generating plants in the New England states. May through July, 1996

Investigated whether the December 25, 1993, turbine generator failure and fire at the Fermi 2 generating plant was caused by Detroit Edison Company's mismanagement of fabrication, operation or maintenance. 1995. Client was the Attorney General of the State of Michigan.

Investigated whether the outages of the two units at the South Texas Nuclear Generating Station during the years 1990 through 1994 were caused or extended by mismanagement. Client was the Texas Office of Public Utility Counsel.

Assisted the City Public Service Board of San Antonio, Texas in litigation over Houston Lighting & Power Company's management of operations of the South Texas Nuclear Generating Station.

Investigated whether outages of the Millstone nuclear units during the years 1991 through 1994 were caused or extended by mismanagement. Client was the Office of the Attorney General of the Commonwealth of Massachusetts.

Evaluated the 1994 Decommissioning Cost Estimate for the Maine Yankee Nuclear Plant. Client was the Public Advocate of the State of Maine.

Evaluated the 1994 Decommissioning Cost Estimate for the Seabrook Nuclear Plant. Clients were investment firms that were evaluating whether to purchase the Great Bay Power Company, one of Seabrook's minority owners.

Investigated whether a proposed natural-gas fired generating facility was need to ensure adequate levels of system reliability. Examined the potential impacts of environmental regulations on the unit's expected construction cost and schedule. 1992. Client was the New Jersey Rate Counsel.

Investigated whether Public Service Company of New Mexico management had adequately disclosed to potential investors the risk that it would be unable to market its excess generating capacity. Clients were individual shareholders of Public Service Company of New Mexico.

Investigated whether the Seabrook Nuclear Plant was prudently designed and constructed. 1989. Clients were the Connecticut Office of Consumer Counsel and the Attorney General of the State of Connecticut.

Investigated whether Carolina Power & Light Company had prudently managed the design and construction of the Harris nuclear plant. 1988-1989. Clients were the North Carolina Electric Municipal Power Agency and the City of Fayetteville, North Carolina.

Investigated whether the Grand Gulf nuclear plant had been prudently designed and constructed. 1988. Client was the Arkansas Public Service Commission.

Reviewed the financial incentive program proposed by the New York State Public Service Commission to improve nuclear power plant safety. 1987. Client was the New York State Consumer Protection Board.

Reviewed the construction cost and schedule of the Hope Creek Nuclear Generating Station. 1986-1987. Client was the New Jersey Rate Counsel.

Reviewed the operating performance of the Fort St. Vrain Nuclear Plant. 1985. Client was the Colorado Office of Consumer Counsel.

WORK HISTORY

2000 - Present: Senior Consultant, Synapse Energy Economics, Inc.

1994 - 2000: President, Schlissel Technical Consulting, Inc.

1983 - 1994: Director, Schlissel Engineering Associates

1979 - 1983: Private Legal and Consulting Practice

1975 - 1979: Attorney, New York State Consumer Protection Board

1973 - 1975: Staff Attorney, Georgia Power Project

EDUCATION

1983-1985: Massachusetts Institute of Technology
Special Graduate Student in Nuclear Engineering and Project Management,

1973: Stanford Law School,
Juris Doctor

1969: Stanford University
Master of Science in Astronautical Engineering,

1968: Massachusetts Institute of Technology
Bachelor of Science in Astronautical Engineering,

PROFESSIONAL MEMBERSHIPS

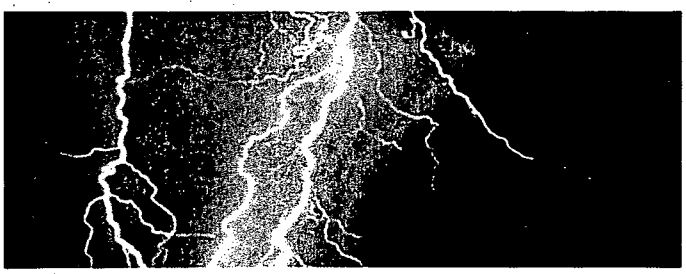
- New York State Bar since 1981
- American Nuclear Society
- National Association of Corrosion Engineers

Synapse
Energy Economics, Inc.

Report on the Availability of Replacement Capacity and Energy for Indian Point Units 2 & 3

November 28, 2007

AUTHOR
David Schlissel



22 Pearl Street
Cambridge, MA 02139
www.synapse-energy.com
617.661.3248

Entergy Nuclear Operations has applied to the Nuclear Regulatory Commission for a renewal the two operating licenses for Indian Point Unit 2 and Unit 3 for an additional 20 years. This report examines the availability of: (1) renewable energy resources, (2) energy conservation and efficiency measures, (3) repowering of existing power plants, (4) transmission system upgrades and enhancements and (5) new power plants. The report concludes that the capacity and energy provided by Indian Point Units 2 and 3 can be replaced if the Units are not relicensed. In particular, energy efficiency, renewable resources, the repowering of older generating facilities, transmission upgrades and new natural gas-fired generating facilities represent viable alternatives to the relicensing of Indian Point. Substantial reductions in peak demand and energy requirements will be achieved by 2013 under the state's newly announced "15 by 15" Clean Energy Plan. Significant amounts of new renewable resources will be available as a result of the state's renewable energy portfolio standard and other initiatives. In addition, thousands of megawatts ("MW") of new generating capacity can be provided by the repowering (i.e., rebuilding) of older generating facilities both along the Hudson River and in the downstate area of the state in New York City and on Long Island. At the same time, transmission system upgrades also can increase the amounts of power that can provided to the downstate region of the State. Finally, there is the potential for the addition of several thousand megawatts of new generating facilities in the Hudson River Valley and in downstate New York.

This report was prepared by David A. Schlissel. Mr. Schlissel is a Senior Consultant at Synapse Energy Economics. Since 1973, he has served as a consultant, expert witness, and attorney on complex management, engineering, and economic issues, primarily in the fields of energy and the environment. Prior to joining Synapse, Mr. Schlissel was the president of Schlissel Technical Consulting, Inc. and its predecessor, Schlissel Engineering Associates.

Mr. Schlissel has been retained by regulatory commissions, consumer advocates, publicly-owned utilities, non-utility generators, governmental agencies, and private organizations in 23 states to prepare expert analyses on issues related to electric, natural gas, and telephone utilities. He has presented testimony in more than 100 cases before regulatory boards and commissions in 28 states, two federal regulatory agencies, and in state and federal court proceedings.

Recent work has involved the evaluation of electric transmission and distribution system reliability, power plant operations and outages, industry restructuring including quantification of stranded costs, proposed nuclear and fossil power plant sales, and proposed utility mergers. Mr. Schlissel has also examined the impact of nuclear power plant life extension on plant decommissioning costs.

Mr. Schlissel holds BS and MS degrees in Astronautical Engineering from the Massachusetts Institute of Technology (MIT) and Stanford University. He also received a Juris Doctor degree from Stanford University School of Law. He has also studied Nuclear Engineering and Project Management at MIT. He is a member

of the New York State Bar, the National Association of Corrosion Engineers, and the American Nuclear Society.

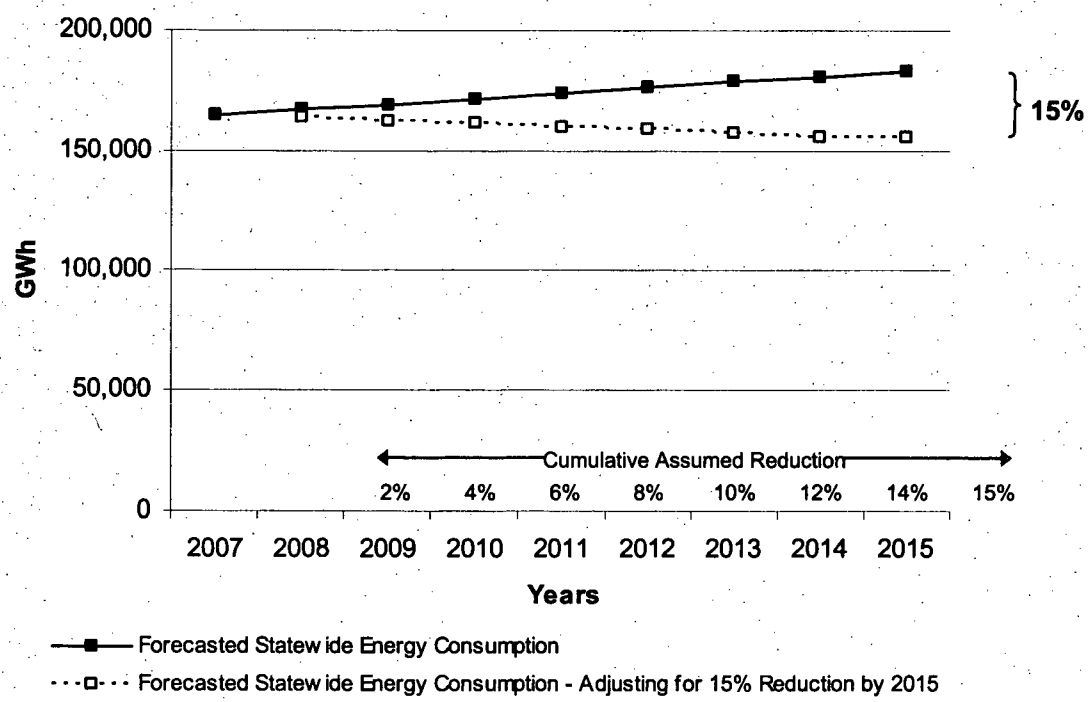
ENERGY EFFICIENCY

New York Governor Eliot Spitzer has announced a "15 by 15" Clean Energy Plan to reduce energy consumption in 2015 by 15 percent to be achieved by energy efficiency alone.[1] The energy efficiency that would be achieved under this Plan would more than replace the capacity and energy provided by both Indian Point Units.

As explained by the Governor, the plan would include taking actions to provide incentives to utilities to conserve energy, strengthening efficiency standards for energy intensive appliances and buildings, and by making the State Government's use of energy more efficient.

The "15 by 15" plan would reduce statewide electricity consumption by approximately 27,000 GWh by 2015. Figure 1 below illustrates the energy savings that would be achieved under the program assuming a linear implementation.

Figure 1 – Impact of New York State's "15 by 15" Policy



The reasonably expected annual generation from both Indian Point Units after 2013 would be approximately 15,600 GWh. This reflects a capacity rating of 979MW for

[1] Remarks by Governor Eliot Spitzer. "15 by 15": A Clean Energy Strategy for New York. 19 Apr 2007. Found at: http://www.state.ny.us/governor/keydocs/0419071_speech.html

Indian Point Unit 2, a 1,000 MW capacity rating for Indian Point Unit 3, and 90 percent average annual capacity factors for both units. The capacity ratings for each unit reflect approximately 4 percent reductions in net plant output due to the potential addition of cooling towers.

To determine the potential of this policy to offset the Indian Point units, we evaluated the potential energy and summer peak capacity savings that can be expected from the "15 by 15" policy using both statewide[2] and zonal[3] forecasts of energy consumption in GWh by the New York Independent System Operator (NY ISO). We used zonal forecasts from Zones H, I, J and K to represent the region that the Indian Point units directly serve. However, it is also relevant to look at the potential for summer peak capacity savings statewide as the region does import power from other regions.

The ramp-in required to achieve the target of 15% energy reduction by 2015 had not been determined. Therefore, we assumed a linear ramp-in of 2% per year starting in 2008 and ending in 2014, with 1% remaining required in 2015 to reach the goal of 15%. We calculated the statewide and regional energy reductions that would be required to achieve this goal by multiplying the total forecasted energy consumption by state and region by the cumulative percentage reduction required for the given year.

We assumed that only 15% reductions would be achieved in the regions of New York State directly served by Indian Points (i.e., Zones H, I, J and K). This is a conservative assumption because it is likely that urban areas such as New York City and Long Island would be able to achieve greater energy reductions than more rural areas which would have fewer energy savings opportunities.

We then converted the energy reductions to summer peak capacity savings in order to assess the ability for these reduction goals to offset the need for the two Indian Point units after 2013. We calculated a ratio between summer peak capacity and energy based on achievable potential estimates from the most recent study of energy efficiency potential in New York State. This study was conducted for NYSERDA in 2003 by Optimal Energy Inc.[4]

Statewide

We used the following methodology to develop ratios to be applied to estimated statewide energy reductions. As zones in the state have a range of avoided costs, I calculated a range of summer peak capacity savings using low and high avoided cost scenarios.

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2 New York Independent System Operator (NY ISO). 2007 Load and Capacity Data. Table I-1. NYISO 2007 Long Term Forecast - 2007 to 2017: Energy (GWh). Pg. 4.

3 New York Independent System Operator (NY ISO). 2007 Load and Capacity Data. Table I-2a. Forecast of Annual Energy by Zone – GWh. Pg. 5.

4 Optimal Energy, Inc. Energy Efficiency and Renewable Energy Resource Development Potential in New York State. Final Report. Volume One: Summary Report. August 2003. Found at: http://www.nyserda.org/sep/EE&ERpotentialVolume1.pdf

Low Avoided Cost Scenario: A ratio between the statewide economic potential summer peak capacity savings and energy reductions in 2007 using low avoided costs (0.196) was applied to energy reductions from 2007-2011. A ratio between the statewide economic potential summer peak capacity savings and energy reductions in 2012 using high avoided costs (0.216) was applied to energy reductions from 2012-2015.[5]

High Avoided Cost Scenario: A ratio between the statewide economic potential summer peak capacity savings and energy reductions in 2007 using high avoided costs (0.212) was applied to energy reductions from 2007-2011. A ratio between the statewide economic potential summer peak capacity savings and energy reductions in 2012 using high avoided costs (0.229) was applied to energy reductions from 2012-2015.[6]

The energy reductions were multiplied by these ratios to arrive at summer peak capacity savings. A summary of the statewide results are shown in Table 1 below.

Table 1 – Statewide Summer Peak Capacity Savings Under "15 by 15"

	Cumulative Energy Reduction (GWh)	Cumulative Summer Peak Capacity Savings – Range from Low to High Avoided Costs (MW)	Indian Point – Cumulative Capacity (MW)
2008	3,349	656 – 710	
2009	6,779	1,328 – 1,436	
2010	10,305	2,019 – 2,183	
2011	13,923	2,728 – 2,950	
2012	17,662	3,817 – 4,049	
2013	21,451	**4,636 – 4,918**	**979**
2014	25,358	5,480 – 5,813	
2015	27,532	**5,950 – 6,311**	
2016			**1,979**

It is clear from this analysis that a statewide 15% energy reduction by 2015 would more than offset the power that would be provided by the two Indian Point units if they were relicensed.

5 Optimal Energy, Inc. Energy Efficiency and Renewable Energy Resource Development Potential in New York State. Final Report. Volume One: Summary Report. August 2003. Table 1.5 New York Statewide Economic Potential – Low Avoided Costs. Pg. 3-4. Found at: http://www.nyserda.org/sep/EE&ERpotentialVolume1.pdf

6 Optimal Energy, Inc. Energy Efficiency and Renewable Energy Resource Development Potential in New York State. Final Report. Volume One: Summary Report. August 2003. Table 1.6 New York Statewide Economic Potential – High Avoided Costs. Pg. 3-4. Found at: http://www.nyserda.org/sep/EE&ERpotentialVolume1.pdf

The Need for Indian Point in Downstate New York (Zones H, I, J and K)

We used a similar methodology to develop ratios to be applied to estimated regional energy reductions (including Zones H, I, J and K). The only difference was that we used higher avoided costs rather than the range of avoided costs to represent these zones because these zones typically have the highest avoided costs in the state.

Table 2 – Regional Summer Peak Capacity Savings Under "15 by 15"

	Cumulative Energy Reduction (GWh)	Cumulative Summer Peak Capacity Savings – High Avoided Costs (MW)	Indian Point – Cumulative Capacity (MW)
2008	1,748	370	
2009	3,541	750	
2010	5,394	1,143	
2011	7,301	1,547	
2012	9,288	2,129	
2013	11,282	**2,586**	**979**
2014	13,340	3,058	
2015	14,487	**3,321**	
2016			**1,979**

Again, a 15% energy reduction in 2015 statewide would more than offset both the energy and capacity from both Indian Point units and would eliminate any need to extend the license of the two units in 2013 and 2016.

Significantly, the 15 percent reduction in statewide energy consumption anticipated under the "15 by 15" plan would not represent all of the economical potential energy efficiency that has been identified in New York State. A recent presentation by Philip Mosenthal of Optimal Energy, Inc., has projected that there is 61,506 GWh of economically potential energy efficiency in the State.[7]

[7] *Electric & Natural Gas Efficiency Potential in New* York, presentation by Philip Mosenthal, Optimal Energy, Inc., at the New York State Public Service Commission Energy Efficiency Portfolio Standard Overview Forum, July 19, 2007, slide no. 9.

RENEWABLE ENERGY RESOURCES

According to NYSERDA's August 2007 *New York State Renewable Portfolio Standard Performance Report for the Program Period ending March 2007,* new renewable capacity installed since the onset of the Renewable Portfolio Standard (RPS) program could exceed 1,206 MW by the end of 2008, of which 1,184 MW would be located in New York State (p.2). The 1,206 MW of new installed capacity is expected to produce approximately 3.6 million MWh of electricity per year.[8]

This same Performance Report also noted that the September 24, 2004 New York PSC Order set forth annual energy targets representing how much renewable energy should be used by New York ratepayers to satisfy the 2013 goal of having 25% of the power consumed in New York come from renewable energy. The RPS energy targets set by the PSC in its September 24, 2004 Order are shown in Table 3 below.

Table 3 - RPS Energy Targets Set by New York Public Service Commission

	Main Tier Targets	Customer Sited Tier Targets	EO 111 Targets	Voluntary Market Targets	Combined Targets
2006	1,121,247	25,259	282,812	228,584	1,657,902
2007	2,326,171	50,488	314,579	457,167	3,148,405
2008	3,549,026	75,685	346,366	685,751	4,656,828
2009	4,767,994	100,855	378,174	914,335	6,161,358
2010	6,012,179	125,988	410,002	1,142,919	7,691,088
2011	7,297,746	151,081	391,857	1,371,502	9,212,186
2012	8,556,710	176,123	373,712	1,600,086	10,706,631
2013	9,854,038	201,130	355,568	1,828,670	12,239,406

Note: Not shown are energy targets associated with voluntary compliance by the Long Island Power Authority (LIPA) and the New York Power Authority (NYPA)

To meet these targets, New York will require the addition of the following MW of renewable resources:

Table 4 - Estimated Renewable Energy Capacity for NY through 2013

Type	Capacity (MW)
Co-fire biomass	296
Hydro	1,100
LFG	121
Off-shore wind	579
Wind	2,450
Solar	16
Small wind	1
Fuel cell	28
Grand Total	4,590

[8] At page 1.

There are an increasing number of analyses of the potential for renewable resources in New York State. It is reasonable to expect that the retirement of either or both Indian Point units at the end of their current NRC licenses would provide a substantial impetus to the development of additional renewable resources.

Wind Powering America: New York, a website sponsored by the US DOE, estimates that the in-state wind energy potential for New York State is 8,400 MW of capacity after land use and environmental exclusions. (available at http://www.nrel.gov/docs/fy00osti/28090.pdf).

NYISO's September 2007 *Comprehensive Reliability Plan 2007*, noted the following concerning wind capacity:

> The NYISO interconnection queue includes proposals for wind generation that now total in excess of 5,000 MW. Wind generators are intermittent resources and have unique electrical characteristics that pose challenges for planning and operations of the interconnected system. The NYISO has completed a study conducted with GE Energy which evaluated the reliability and operating implications of the large scale integration of wind generation. The study concluded that if state-of-the-art wind technology is utilized, wind generation can reliably interconnect with only minor adjustments to existing planning, operating, and reliability practices.[9]

The study cited in this NYISO report is titled *The Effects of Integrating Wind Power on Transmission System Planning, Reliability, and Operations*, *Report on Phase 1, Preliminary Overall Reliability* Assessment, prepared for NYSERDA by GE Energy Consulting, 2004. A Phase 2 Report, *System Performance Evaluation,* also was completed in March 2005.

When combined with other energy resources, wind can produce energy in patterns comparable to a baseload generation facility. At the same time, the effects of short term wind variability can be mitigated by building a larger number of wind turbines and by siting the wind turbines in different geographic locations. There is no evidence that any replacement capacity for Indian Point would need to be a fully dispatchable facility. Indeed, the electric grid in New York State will already have a large number of fully dispatchable facilities.

Entergy merely rehashes the same tired old arguments against reliance on wind power. As a detailed 2004 Wind Integration Study – Final Report prepared for Xcel Energy and the Minnesota Department of Commerce has noted:

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9 New York Independent System Operator, *The Comprehensive Reliability Plan 2007, A Long-Term ReliabilityAssessment of New York's Bulk Power System,* Final Report, September 2007, Appendices, at page 75.

> Many of the earlier concerns and issues related to the possible impacts of large wind generation facilities on the transmission grid have been shown to be exaggerated or unfounded by a growing body of research studies and empirical understanding gained from the installation and operation of over 6000 MW of wind generation in the United States.[10]

Contrary to what Entergy has claimed, wind power can reduce the need for the capacity from Indian Point Units 2 and 3 and can provide low cost energy.

An August 2003 study prepared for NYSERDA, *Energy Efficiency and Renewable Energy Resource Development Potential in New York State,* by Optimal Energy, Inc., American Council for an Energy Efficient Economy, the Vermont Energy Investment Corporation and Christine T. Donovan Associates, has provided the following estimates of the potential for renewable resources and energy efficiency in New York State:

Table 5 - New York Statewide Economic Potential – Low Avoided Costs

	2007		2012		2022	
	Annual GWh	Summer Peak MW	Annual GWh	Summer Peak MW	Annual GWh	Summer Peak MW
Energy Efficiency Savings						
Residential	10,124	1,475	12,205	1,981	15,610	2,646
Commercial	27,490	6,173	32,124	8,009	32,994	9,266
Industrial	5,718	840	6,045	896	4,999	752
Total Efficiency	43,332	8,489	50,374	10,886	53,603	12,664
Renewable Supply						
Biomass	5,141	833	5,325	861	6,344	1,022
Fuel Cells	-	-	-	-	-	-
Hydropower	1,512	109	4,336	375	9,123	816
Landfill Gas	-	-	-	-	-	-
Municipal Solid Waste	-	-	682	91	1,421	190
Photovoltaics	-	-	-	-	-	-
Solar Thermal	175	-	181	-	189	-
Windpower	-	-	1,245	100	41,818	3,255
Total Renewable	6,828	942	11,769	1,427	58,894	5,283
Total Efficiency Savings & Renewable Supply	50,159	9,431	62,143	12,313	112,497	17,947

10 *Wind Integration Study-Final Report*, prepared for Xcel Energy and the Minnesota Department of Commerce by EnerNex Corporation and Wind Logics, Inc., dated September 28, 2004, the Project Summary portion of which is included as Exhibit JI-4-A, at page 19.

Table 6 - New York Statewide Economic Potential - High Avoided Costs

	2007		2012		2022	
	Annual GWh	Summer Peak MW	Annual GWh	Summer Peak MW	Annual GWh	Summer Peak MW
Energy Efficiency Savings						
Residential	12,593	2,433	15,982	3,267	19,660	4,480
Commercial	30,273	7,021	35,340	8,988	36,847	10,225
Industrial	5,718	840	6,045	896	4,999	752
Total Efficiency	48,584	10,294	57,367	13,151	61,506	15,457
Renewable Supply						
Biomass	5,141	833	5,325	861	6,344	1,022
Fuel Cells	-	-	-	-	-	-
Hydropower	2,115	257	5,038	555	10,311	1,095
Landfill Gas	439	59	407	54	419	56
Municipal Solid Waste	-	-	682	91	1,421	190
Photovoltaics	-	-	-	-	-	-
Solar Thermal	175	-	181	-	189	-
Windpower	893	70	3,744	293	41,818	3,255
Total Renewable	8,762	1,219	15,376	1,855	60,501	5,618
Total Efficiency Savings & Renewable Supply	57,347	11,513	72,744	15,006	122,007	21,074

Based on the results of this study, renewable resources have the technical and economic potential to provide between 1427 MW and 1855 MW of new capacity in New York State by 2012 and between 5283 MW and 5618 MW of new capacity by 2022. Energy Efficiency and renewable resources together have the technical and economic potential to provide between 12,313 MW and 15,006 MW in 2012 and between 17947 MW and 21074 MW in 2022. Clearly, this is far more than would be required to replace the approximately 2000 MW of capacity from Indian Point Units 2 and 3.[11]

The same conclusion is true for the energy that would be supplied by Indian Point Units 2 and 3 if their licenses are renewed. The same tables presented above show that renewable resources, alone have the potential to provide between 11769 and 15376 GWh of energy in 2012 and between 58894 and 60501 GWh of energy in 2022. Similarly, energy efficiency and renewable resources combined could provide between 62,143 GWh and 72,744 GWh in 2012 and between 112,497 GWh and 122,007 GWh in 2022.[12]

The 2003 study for NYSERDA also showed that a significant portion of the energy that could be provided by energy efficiency and renewable resources would be in downstate New York.[13] For example, the study found that by 2012, energy efficiency and renewable resources have a technical and economic potential of

11 At Volume One, page 3-4.

12 Id.

13 Id, Figure 1.8, at page 3-7.

approximately 30,000 GWh just in Zones J and K, which represent New York City and Long Island. It similarly found that by 2022, energy efficiency and renewable resources have a technical and economic potential of more than 50,000 GWh just in these same areas of the state. Again, this would easily replace the energy that would be provided by Indian Point Units 2 and 3.

The May 2007 study, *New York's Solar Roadmap, A Plan for Energy Reliability, Security, Environmental Responsibility and Economic Development in New York State*[14], has noted that a private-sector initiative launched in 2007 R&D, manufacturing, and industry leaders in New York State, has developed the strategic goal of increasing solar power deployment in the State from the current level of about 12 MW of grid-connected electricity as of January 2007 to over 2,000 MW by 2017.[15] This would provide about 5 percent of the peak electric capacity of the state.[16]

An October 2002 study by NYSERDA on *Combined Heat and Power, Market Potential for New York State*, has concluded that by 2012 there could be between 763.6 MW and 2,169.1 MW of combined heat and power in the state.[17] Between 525.4 MW and 1,319.7 MW of this combined heat and power could be in the Downstate area of the State.[18]

The new administration in New York State already is taking significant actions to increase the amount of energy efficiency and renewable resources:

New York State has announced the following major initiatives as part of their Clean Energy Agenda:

- **Reduce energy consumption.** Governor Spitzer has announced that New York will reduce energy consumption by 15 percent below the forecasted level in 2015 – this is the most aggressive target in the country. New York businesses can raise their profits and New York's families can reduce their utility bills by conserving energy. At the state level, government will lead by example and cut its own use of energy.

- **Invest in and develop renewable energy such as wind, solar, hydropower, and fuel cells**. The Spitzer-Paterson administration will ensure New York will meet the current goal of obtaining 25 percent of our energy from renewable resources by 2013, and the Task Force will evaluate whether to expand this goal. In addition,

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14 This study is available at http://www.neny.org/download.cfm/NENY_Membership_Application.pdf?AssetID=225

15 Executive Summary, at page 1.

16 Id, at page 2.

17 *Combined Heat and Power, Market Potential for New York State*, NYSERDA, Final Report 02-12, October 2002, Table ES-4, at page ES-9.

18 Id.

we must continue to support research and development in this area, and encourage renewable energy businesses to locate in New York.

- **Clean Energy Siting Bill.** Streamlining the state approval process for renewable and clean energy sources is an essential part of our effort. Governor Spitzer proposed a new power plant siting law ("Article X") that would provide a streamlined and expedited review process for wind projects and other clean energy sources. [19]

The State also has convened a Renewable Energy Task Force to evaluate, among other issues, whether the state's Renewable Portfolio Standard should be increased to 30 percent as a result of the Governor's announced "15 by 15" energy efficiency program.

POWER PLANT REPOWERING

Entergy did not consider the potential repowering of older existing power plants as an alternative to the relicensing of Indian Point Units 2 and 3.

Repowering a generation facility means replacing a plant's old, inefficient and polluting equipment with newer, more efficient equipment. Today, virtually all repowering projects replace old equipment with combined-cycle combustion turbines (CCCTs). CCCTs generate electricity in two stages. In the first stage, fuel is burned to operate a gas turbine generator, and in the second stage, excess heat from the gas turbine is used to drive a steam turbine and generate additional electricity. This two-stage process can turn 50 percent or more of the fuel energy into electricity. Repowering has become commonplace in the electric industry since the early 1990s. One repowering project in the Hudson River Valley was PSEG's Bethlehem Energy Center outside Albany. Completed in 2005, this project now consists of 793 MW of combined-cycle generating capacity, which includes a net increase of 400 MW relative to the old Albany Steam Plant that was replaced.

In practice, repowering can be done in at least two ways, either by rebuilding and replacing part or all of an existing plant or by closing down an existing power plant, building a new unit next to it and reusing the existing transmission and fuel facilities.

Repowering older power plants provides a number of important environmental and electric system reliability benefits: improved plant availability, lower plant operating and maintenance costs; increased plant capacity and generation; reduced facility heat rates which lead to significantly more efficient fuel use; reuse of industrial sites; up to 99 percent reductions in water intake and related fish impacts; and large reductions in air emissions, both overall and in terms of emissions per MWh of electricity.

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[19] Available from http://www.ny.gov/governor/press/lt_conservation.html.

A recent study on repowering KeySpan's generating facilities on Long Island by the Center for Management Analysis at Long Island University concluded that repowering these facilities would provide cost effective generating capacity to carry Long Island at least into the next 20 to 40 years and beyond, and would provide "compelling" environmental benefits:

> Improvements in efficiency from about 35 percent to close to 60 percent in the conversion of fuel to electricity can be achieved. The resulting reduction in fuel burned for a given amount of generation will be significantly less nitrogen oxides and carbon monoxide emitted. Modern combined cycle units have state of the art emission control systems in contrast to the older steam electric units with no such controls. The re-powered units achieve emission reductions immediately since they replace higher emitting, older units that would likely continue to operate in an expansion program of new greenfield projects.[20]

The study by the Center for Management Analysis concluded that converting the major plants on the KeySpan system to combined cycle could increase Long Island's electric supply by about 2,000 MW.[21] Clearly, the repowering of these existing power plants on Long Island could replace the approximate 2,000 MW of capacity provided by Indian Point Units 2 and 3.

Reliant Energy also received an Article X certificate to repower its aging Astoria Generating facility. This repowering would add another 1,816 MW of combined cycle capacity to the electric system in New York City. This would represent an increase of approximately 650 MW over the capacity of the existing Astoria facility. The retirement of Indian Point Units 2 and 3 would create an incentive for the completion of this repowering project.

Detailed engineering and economic analyses must be performed to determine the optimum size of the repowered unit and the extent to which existing facilities can be refurbished and reused. The types of existing facilities that can be refurbished and reused include boilers, turbine generators, condensers, transmission switchyards, and other auxiliary plant equipment. The reuse of this equipment can lower the cost of building the repowered facility as compared to the cost of constructing a new unit at a new site.

There are a number of older fossil-fueled power plants situated on the river between Albany and New York City: Bowline Point, Roseton, and Danskammer. As noted earlier, one older plant along the river, the old Albany Station, has been replaced with modern power generation equipment. However, the units at the Bowline, Roseton and Danskammer fossil-fueled plants utilize older power generating technology, which is less efficient and has far greater environmental impacts than new generating systems. Most of the boilers and generating units in these four plants are over 25 years old – three of them are over 45 years old – and none of them has been retrofitted with post-combustion emission controls or modern

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20 *The Feasibility of Re-Powering KeySpan's Long Island Electric Generating Plants to Meet Future Energy Needs*, Long Island University, Center for Management Analysis, August 6, 2002, at page 8.

21 Id, at page 78.

cooling systems that minimize water use from the river. Repowering these plants with new combined cycle technology could add additional generating capacity to replace Indian Point at the same time that it would provide significant economic and environmental benefits.

TRANSMISSION SYSTEM ENHANCEMENTS AND UPGRADES

Entergy has failed to adequately consider transmission system enhancements and upgrades as part of the portfolio of options for replacing the capacity and energy from Indian Point Units 2 and 3. Such enhancements and upgrades could increase the capability to import power into the Hudson River Valley and Downstate New York from New England, PJM[22] or upstate New York.

For example, at least two new transmission links between New York and New Jersey have been proposed. Both of these are in the interconnection queue at the New York ISO. One of these is the Hudson Transmission Project that would provide a new controllable line into New York City rated at 600 MW.[23] A second project, the 550 MW Harbor Cable Project and Generating Portfolio, would provide a full controllable transmission pathway from generating sources in New Jersey to New York City.[24]

At the same time, the 2005 Levitan & Associates study identified three possible transmission alternatives to the retirement of Indian Point Units 2 and 3. The first would include retirement with the construction of two physically separate 500 kV circuits between the Capitol District around Albany to the downstate grid in New York City. Each of the circuits would be controllable and would be able to transmission 1,000 MW of power for a total of 2,000 MW.[25] A third proposed project would be the 300 MW Linden Variable Frequency Transformers that would be physically located adjacent to the Linden Cogen plant in northern New Jersey. It would result in a variable 300 MW tie between PJM and New York City.[26]

The second transmission alternative identified by Levitan & Associates would be to upgrade the existing 345 kV New Scotland-Leeds circuit and the 345 kV Leeds-Pleasant Valley circuit, and construct a new 345 kV line from New Scotland to Pleasant Valley. This would increase the UPNY-SENY interface transfer capability by approximately 600 MW.[27]

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22 PJM is the interconnected regional electric system in 13 states and the District of Columbia. New Jersey and Pennsylvania are two of the state's within PJM.

23 New York Independent System Operator, *The Comprehensive Reliability Plan 2007, A Long-Term ReliabilityAssessment of New York's Bulk Power System,* Final Report, September 2007, at page 27.

24 Id.

25 *Indian Point Retirement Options, Replacement Generation, Decommissioning/Spent Fuel Issues, and Local Economic/Rate Impacts*, prepared for the County of Westchester and the County of Westchester Public Utility Service Agency, by Levitan & Associates, Inc., June 9, 2005, at pages 35 and 36.

26 Id.

27 Id, at pages 36 and 37.

Finally, the third transmission alternative would be to convert the existing 345 kV Marcy-New Scotland circuit to a double circuit and to rebuild the New Scotland station to a breaker-and-a-half design. This would increase the Central-East transfer capability by approximately 650 MW and increase the transmission capability into New York City by approximately 450 MW.[28]

Levitan & Associates also identified a fourth transmission alternative that would upgrade the interconnections between New York and the PJM system by re-conductoring the existing transmission paths from Ramapo to Buchanan and/or constructing a new dedicated (overhead or underground) transmission line from Ramapo to Buchanan. However, Levitan & Associates were unsure of the amount by which this alternative would increase the Total East transfer capability into New York State.

NEW GENERATING FACILITIES

A number of proposed power plant projects received certificates under New York's now-expired Article X statutes. However, some of these projects have not been built because they were unable to secure the needed financing. The Governor of New York has proposed requiring utilities to enter into long-term contracts with prospective suppliers. This would enable plant developers to limit risks, gain the confidence of investors and obtain the financing to build their projects.

The following is list of the approved projects in the Hudson River Valley and downstate New York that have not been built:

- Besicorp – Empire State Newsprint Project – 505 MW – Rensselaer County
- Bowline Unit 3 – 750 MW – Rockland County
- Reliant Energy Astoria Repowering Project – 1816 MW total (net addition 652 MW) – Queens County
- Spagnoli Road Energy Center – 250 MW – Suffolk County

The addition of these units would add over 2,100 MW of new generating capacity.

Other new generating facilities, totaling 1400 MW of new capacity, have been proposed for downstate New York including:

- A second Astoria Repowering Project, submitted by NRG Power Marketing, would add 500 MW (375 MW net) of new combustion turbine power in Queens by 2011.[29]
- A 600 MW combined cycle unit at Arthur Kill on Long Island by 2012.[30]

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28 Id, at page 37.

29 New York Independent System Operator, *The Comprehensive Reliability Plan 2007, A Long-Term Reliability Assessment of New York's Bulk Power System,* Final Report, September 2007, at page 27.

30 Id.

- A 300 MW Peaking Facility at Indian Point, proposed by Entergy Nuclear Power Marketing. This project would be in service by mid-2011.

As explained in the 2005 *Indian Point Options* study by Levitan & Associates, it is reasonable to expect that the retirement of Indian Point would encourage developers to complete the approved but not yet built projects:

> Project developers are keenly tuned to market dynamics in New York. They would realize that retiring IP would cause market energy and capacity values to increase across the downstate region. These price signals would be important, given IP's size and location, to encourage the development of new generation and/or transmission projects that would replace the lost capacity. These new generation projects could include decentralized and renewable resource options. If the retirement of IP were announced in advance, developers would be able to calculate the economic feasibility of their projects and pursue those that make financial sense in time to maintain the state's reliability requirement. In addition, utilities in the downstate regions might offer long-term PPAs for new replacement generation. PPAs offer generators market certainty and reduce price risk, improving the opportunity for owners to obtain debt and equity financing in today's skittish financial markets.
>
> The developers' ability to respond to market price signals and the utilities' interest in contracting for new generation are central to our analysis. We believe that developers would require a minimum of three-to-four years to plan, permit, and construct a gas-fired combined cycle project. Perhaps six months to a year could be shaved off the time for a simple cycle project. The early project development work can often be accomplished at minimal cost, even if a formal retirement plan was not announced, in order for the developer to get a "head start" on competitors. Such tasks encompass conceptual design, site control, preliminary fuel supply and power offtake arrangements, and initial permit applications. The remaining project development and construction time would be approximately three years for a combined cycle plant and less for simply cycle. Thus we would recommend that any voluntary retirement be announced at least three-to-four years in advance, to give the market enough time to develop replacement capacity....
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> * * * *
>
> The existing NRC license expiration dates of 2013/15 define our Base Case scenario against which we evaluate other options. If Entergy announced an agreement to retire IP2&3 on those dates at least three, and preferably four years in advance, there would be

more than enough time for project developers and downstate utilities to respond.[31]

It is important to realize that gas supply will not be a critical factor in closing Indian Point. According to the 2006 National Academy of Sciences study, " *Committee on Alternatives to Indian Point for Meeting Energy Needs*, at page 5, replacing both Indian Point units would ultimately require an additional 1300-1400 MW of new gas-fired generating capacity. Conservatively assuming a heat rate of 8000 btu/KWh, under peak conditions providing 1400 MW would require a gas supply of 0.26 bcf per day, or about 16% of the combined capacity of the new LNG facilities being developed in Eastern Canada and Massachusetts. There will be more than enough slack in the system to supply the gas needed for additional generating facilities to replace Indian Point from existing and new sources outside New York State.

New gas supplies will be available in the northeastern United States and eastern Canada from new LNG facilities that are expected to be on-line within the next few years. (The Canaport LNG terminal is expected to begin receiving deliveries and transporting gas to the northeast United States through the upgraded Maritimes and Northeast pipeline as soon as 2008) The combined capacity of these LNG terminals would be approximately 1.73 billion cubic feet (bcf) per day, of which 0.73 bcf would be delivered from the Canaport facility (Nova Scotia) and 1.0 bcf from two offshore facilities in Massachusetts. These facilities are well advanced in the permitting process (Canaport is under construction), and they rely on known and proven LNG transfer and regassification technologies.

Note that the two proposed LNG import terminals, located in Massachusetts, to serve the northeast market have been approved by the Governor of Massachusetts.[32] In addition, the Repsol Energy North America Corporation, developer of the Canaport LNG facility in Saint John, New Brunswick, has filed a notice with FERC clarifying that they intend and expect to deliver 0.73 bcf of gas into the northeastern United States.[33]

The addition of these new LNG facilities in the northeastern United States and eastern Canada will free-up additional pipeline capacity into the New York area from the south so that more gas could be delivered to the Westchester Area. Today, New England gets much of its gas supply from the Algonquin Pipeline which passes through Connecticut from the southeast corner of the state to the northwest corner. This transport—through function accounts for about 90% of the activity on Algonquin in this region. Once additional LNG-based supplies are available in New England, much of that existing pipeline capacity would be available for delivering gas supplies from domestic sources (i.e., the Gulf of Mexico) to the New York area. In addition, decreased competition for this pipeline capacity means that

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31 *Indian Point Retirement Options, Replacement Generation, Decommissioning/Spent Fuel Issues, and Local Economic/Rate Impacts*, prepared for the County of Westchester and the County of Westchester Public Utility Service Agency, by Levitan & Associates, Inc., June 9, 2005, at pages 30 and 31.

32 http://www.boston.com/news/local/articles/2006/12/20/governor_approves_2_lng_ports/

33 http://elibrary.FERC.gov/idmws/file_list.asp?accession_num=20070111-0066

transportation costs to the New York area are likely to decrease. Thus the availability of new LNG terminals in New England and eastern Canada will provide a benefit to New York and Connecticut in terms of availability of supply, and likely in terms of price, even if the physical molecules of gas are not delivered to the region from those new LNG facilities.

In conclusion, the LNG terminals in Canada and Massachusetts will all add to the available gas supplies for New York and Connecticut. They can do this either directly, by transporting gas to the region through the interstate pipeline system, or indirectly, by releasing pipeline capacity that would otherwise be reserved for moving supplies through the region and northward.[34]

CONCLUSION

In conclusion, the capacity and energy provided by Indian Point Units 2 and 3 can be replaced if the Units are not relicensed. In particular, energy efficiency, renewable resources, the repowering of older generating facilities, transmission upgrades and new natural gas-fired generating facilities represent viable alternatives to the relicensing of Indian Point. Substantial reductions in peak demand and energy requirements will be achieved by 2013 under the state's newly announced "15 by 15" Clean Energy Plan. Significant amounts of new renewable resources will be available as a result of the state's renewable energy portfolio standard and other initiatives. In addition, thousands of megawatts ("MW") of new generating capacity can be provided by the repowering (i.e., rebuilding) of older generating facilities both along the Hudson River and in the downstate area of the state in New York City and on Long Island. At the same time, transmission system upgrades also can increase the amounts of power that can provided to the downstate region of the State. Finally, there is the potential for the addition of several thousand megawatts of new generating capacity in the Hudson River Valley and in downstate New York.

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34 See *The Proposed Broadwater LNG Import Terminal: An Analysis and Assessment of* Alternatives, March 2006 and *The Proposed Broadwater LNG Import Terminal Update of Synapse Analysis*, January 19, 2007, both are available at www.synapse-energy.com.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC
Entergy Nuclear Indian Point 3, LLC and
Entergy Nuclear Operations, Inc.

Docket Nos. 50-247-LR and 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

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Declaration of Rudolf H. Hausler

I, Rudolf H. Hausler, Ph.D., declare that the foregoing is true and correct.

1. As the President of Corro-Consulta, Inc., I am employed as an expert to the office of the Attorney General of the State of New York.

2. Corro-Consulta is a sole proprietorship. It has supported the oil industry in both the up-stream and refining areas since 1996 and has at times supported the Nuclear Information and Resource Services (NIRS) and the Union of Concerned Scientists (UCS) with failure analysis and advice. I have also worked with the Electric Power Research Institute (EPRI) on corrosion issues.

3. My educational and professional experience is detailed in the attached curriculum vitae (CV); also attached is a list of my publications, patents, awards, and other professional activities. I received a Master's Degree from the Swiss Federal Institute of Technology in Chemical Engineering in 1958 and a Doctoral Degree from the same institution in Technical Services in 1961.

4. I am making these assessments on the basis of my education and experience as a corrosion engineer with proven expertise (see attached CV for experience and extensive publications) in chemistry, physical chemistry, electrochemistry, corrosion-chemistry, -processes, -mechanisms, and -phenomenology, failure analysis, corrosion modeling and management (inhibitors and other chemical additives), and system analysis for corrosion management.

5. Following catastrophic failures during the early chemical cleaning of the Indian Point 1 steam generator, I worked extensively with the nuclear industry and EPRI in developing and field-testing the only corrosion inhibitor qualified for the cleaning process for the past 25+ years (see EPRI publication 3030, Project S 148-1, final report, 303+ pages).

6. Additionally, I have rendered opinions regarding Davis-Besse reactor vessel cover boric acid corrosion as well as assessments of specific spent fuel dry storage caskets.[1]

7. I am a National Association of Corrosion Engineers (NACE) Certified Corrosion Specialist (life membership 44227-00), have been awarded the NACE Technical Achievement Award, have been elected a NACE Fellow, and am a Professional Engineer - Corrosion Branch (certified in California, certificate No. 258).

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[1]In April of 2002, I supported NIRS in preparing a Petition Pursuant 10 C.F.R. 2.206 regarding safety at Davis-Besse Nuclear Power Plant. I similarly rendered an opinion to The Huntsville Times, Huntsville, AL, regarding cracking of control rod drive mechanism (DRDM) nozzles (Letter to Mr. Brian Lawson, May 29, 2002). These activities required extensive review of the pertinent literature.

8. This declaration represents my current opinion on the topics it covers.

9. I believe that the currently proposed monitoring process for buried pipes at Indian Point 2 and Indian Point 3 as specified in the License Renewal Application is inadequate.

10. I believe that the safety-related piping system at Indian Point 2 and Indian Point 3 is far more deteriorated than is reflected in the License Renewal Application (LRA) and the accompanying Updated Final Safety Analysis Report (UFSAR).

11. I also believe that the proposed monitoring schedule, which envisages an examination cycle of the infrastructure covered by the scope of this declaration of ten years, is inadequate in view of the current age of the structures and their advanced state of deterioration.

12. I further believe that the "preventive measures consisting of maintaining external coatings and wrapping" as stated in the "Aging Management Programs and Activities"[2] is inadequate because it does not address deterioration of the pipes from the inside.

13. I will further demonstrate in this declaration that the proposed "Aging Management Programs" fall far short of standard industry practice, and hence Entergy is unable to demonstrate that its Aging Management Plan provides reasonable assurance of adequate protection of the public health and safety.

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[2]Entergy, License Renewal Application for Indian Point Energy Center, Appendix B, p. B-27, B-42.

14. I believe that my broad experience in corrosion theory, crude oil refineries, oil and gas production, as well as pipeline corrosion, corrosion monitoring, failure analysis and corrosion direct assessment both internally and externally, coupled with a thorough understanding of the theory and application of cathodic protection, make me qualified to render judgment on the subject matter at hand.

Discussion of Internal and External Corrosion Protection, Monitoring, and Maintenance of Buried Pipes at Indian Point Nuclear Generating Units 2 and 3 in Connection with the License Renewal Application

I. Background

15. Entergy Corporation through Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc. (collectively, "Entergy" or "Licensee"), operator of the Indian Point Energy Center, is endeavoring to obtain from the Nuclear Regulatory Commission (NRC) a license to continue operating the power plants Indian Point 2 and Indian Point 3 for another 20 years. The renewed license would in effect extend the operating life of both plants from 40 to 60 years.[3]

16. It is widely understood by Entergy as well as the NRC that structures age, or deteriorate, over time for a variety of reasons and that such aging needs to be managed, particularly in the case of safety-related structures. Specifically, Entergy has submitted an Aging Management Process plan for buried piping as it

[3]However, it is my understanding that the piping system was put in place at Indian Point Units 1, 2 and 3, at least in part, prior to the issuance of the operating licenses for Unit 1, Unit 2 and Unit 3, and that the piping systems have been in the ground for longer than the time period for which the facilities have been in operation.

pertains among others to the Service Water System and Emergency Cooling System. This system is safety-related.[4]

17. The common standard followed by the NRC in issuing an operating license is guided by the following considerations: "that the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively, provide reasonable assurance the applicant will comply with the regulations in the chapter, including the regulations in part 20 of this chapter, and that the health and safety of the public will not be endangered."[5]

18. It is furthermore a requirement that for those structures and components subject to an aging management review the licensee shall "demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB [current licensing basis] for the period of extended operation."[6]

19. The intended function of pipes is not only to carry fluids, but to separate those fluids from the environment, particularly if these same fluids contain radioactive constituents. Similarly, the intended function of tanks is not only to store fluids, but to separate those fluids from the environment, which latter notion is intrinsically imbedded in the notion or meaning of storage.

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[4]*See* LRA, p. A-19, § A.2.1.5, p. B-27, § B.1.6, p. B-42-43, § B.1.11.

[5]10 C.F.R., Chapter 1, pt 50.40 (Standards for Licenses, Certification and Regulatory Approvals).

[6]10 C.F.R., Chapter 1. Pt. 54.213.

II. Evidence of Deterioration of Buried Pipes

20. There have been reports of radioactively contaminated groundwater reaching drinking water wells in the vicinity of nuclear plants. Such occurrences have recently been documented for the Braidwood and Dresden facilities in Illinois.[7]

21. At Indian Point, where wells have been drilled to monitor ground water for radioactive contamination in some cases very high levels of tritium, 250,000 to 300,000 pCi/L have been observed,[8] along with cesium, cobalt, strontium, and nickel.[9]

22. The scarcity of available information concerning buried pipe failures at Indian Point makes assessment of the existing deterioration nearly impossible. By comparison, however, in 2006, Entergy identified 30+ potential leak sources of radioactive material that were readily visible and identifiable from above-ground sources.[10] Of equal or larger concern are those that cannot be detected visually, *i.e.*, those occurring from buried pipes and which are responsible for the massive

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[7]Braidwood Power Reactor Event Number 42184, Offsite Notification - Elevated Levels of Tritium Found in Groundwater (PNO-RIII-05-016A, Dec. 7, 2005), *see also* Associated Press Wire Service, *Tritium Levels Around Braidwood Plant Worry Nuclear Watchdog,* Jan. 25, 2006; Union of Concerned Scientists Fact Sheet: Pipe Leak at Dresden (Oct. 13, 2004); NRC, Preliminary Notification of Event or Unusual Occurrence (PNO-RIII-06-004., Feb. 15, 2006).

[8]E-mail from Donald Croulet, Entergy to Pat Conroy, Entergy, (Jan. 16, 2006): "Executive Summary Sheets for 1/16/06."

[9] *Id.*; Jim Fitzgerald, *High Levels of Strontium-90 Found in Indian Point Groundwater*, Associated Press, Mar. 21, 2006.

[10] E-mail from Donald Croulet of Entergy to James Noggle of USNRC, "regarding H-3 sources IPEC-RL-Comments-1" (attachment, table) (Dec.12, 2005), ML061000598; *see also* E-mail from T.R. Jones of Entergy to Jim Noggle of USNRC, "FW: Definition of Underground liquid services" (Oct. 25, 2005), ML060930102.

contamination of the soil and water table.

III. A History of Failures

23. Aging of pipes by corrosion has been recognized and identified as a serious problem.[11] Corroded pipes recently failed at the Byron Nuclear Plant in Rockford, Illinois.[12] Figure 1, attached, shows the nature and location of the leak.[13] The pipe is specified to have a wall thickness of 0.375 inches and apparently had measured 0.047 inches before the rust was removed, which took the remaining wall thickness to zero. It is not clear at this point in time whether corrosion occurred from the inside or outside of the pipe. In view of the large amount of rust, which was scraped off, one might suspect that corrosion occurred from the outside due to a leaking flange.

24. Local failures at Indian Point have occurred as well, and were described in Entergy's Relief Request regarding a Temporary Repair to a Service Water Pipe.[14] Apparently an essential (safety-related) 18-inch service water line had sprung a leak just downstream of a flange leading into an elbow. This failure

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[11]Risk-Informed Assessment of Degraded Buried Piping Systems in Nuclear Power Plants; Brookhaven National Laboratory; US Nuclear Regulatory Commission, NUREG-6876, June 2005, ML051650146 (hereinafter the "Brookhaven Report").

[12] NRC Preliminary Notification of Event or Unusual Occurrence, PNO-III-07-012, "Both Units at Byron Shut Down Due to a Leak in Pipe," (October 23, 2007) ML072960109. Union of Concerned Scientists, Issue Brief, *Help Wanted: Dutch Boy at Byron* (Oct. 25, 2007).

[13]*Id.*

[14] Entergy Northeast, Indian Point Energy Center, Response to Request for Additional Information Regarding Relief Request 3-43 for Temporary Repair of Service Water Pipe (Oct. 3, 2007), ML072890132.

occurred internally and hence this type of failure could have occurred anywhere in a buried piping system. The area was ultrasonically tested in order to assess the extent of the damage. Figure 3, attached, shows the percentage of residual wall thickness in both the horizontal and vertical direction.[15] Clearly there is a large area where the residual wall thickness is between 30 and 20 percent of original and somewhat smaller areas with only 10 percent or less of the original wall remaining.[16] (I statistically evaluated these measurements as well, see Figure 14, attached.[17]) This failure is typical for flow-induced localized corrosion (also some times erroneously called "erosion corrosion") downstream of a flow upset. No visual representation (photograph) of this failure was available, however Figure 4 (attached) shows for illustration purposes a similar failure observed in the oil field.

IV. Factors Critical to Development of an Aging Management Plan for Buried Pipes

25. Assessment of the potential for corrosion-related failures, and the appropriate aging-management measures that must be taken to address them, must begin with an assessment of current and historical environmental and facility conditions and the history of corrosion. Nowhere in the LRA is there information demonstrating that Entergy has first determined these factors before it reached conclusion stated in the LRA as to corrosion rate or inspection rate. Therefore, discussion of broader types of failure mechanisms, corrosion rate, and consideration of the factors we know to be at play is in order.

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[15] To create this diagram I entered the data points given in Entergy's Oct. 3, 2007 Request for Additional Information into Statistical Analysis Software (JMP program).

[16] Entergy, Response to Request for Additional Information, above note 14.

[17] Above note 15.

A. The Failure Mechanism

26. There are clearly two types of failures: corrosion from the inside and corrosion from the outside. A partial generic list of degradation mechanisms for both internal and external corrosion is detailed below:

a) **General Corrosion** is prevalent on bare metal surfaces in aerated,chloride-containing water at relatively high velocities and near neutral or slightly acidic pH.

b) **Pitting Corrosion** is a localized corrosion phenomenon primarily based on random variation in the many parameters that control the oxidation of the metal to begin with, including defective coating, degraded pipe wrappings, variations in metallurgy, etc.

c) **Crevice Corrosion** is a much misunderstood corrosion phenomenon. Typical crevice corrosion is shown in Figure 11 [18] (attached) and is characterized by a crevice formed by two metal parts such that the environment inside the crevice is starved of oxygen. A subcategory of crevice corrosion is "under-deposit corrosion" where the "crevice" is formed by a non-conducting (or only partially conducting) material. Deposits can be formed by marine life and /microbes (bacteria).

d) **Galvanic Corrosion** is characterized by two dissimilar metals, in intimate contact, and exposed in an electrolyte-forming environment. A weld consists of dissimilar metals. The weld metal is usually dissimilar in composition from the base metal which the heat-affected zone is dissimilar from both in microstructure. It is very difficult to predict which one is going to corrode preferentially to the other. Figures 12 and 13[19] (attached) are typical examples of these phenomena: Figure 12 represents a flow restriction. The resulting turbulence upstream of the weld caused a portion of the heat-affected zone (HAZ) or corrode all the way through. In addition to the weld metal being cathodic (*i.e.*, negative) to the HAZ, the example

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[18] Photograph by R.H. Hausler, circa 1984.

[19] Photographs by R.H. Hausler, circa 1986, St. Louis, Missouri.

also demonstrates that in turbulent situations it is impossible at this stage to predict exactly where corrosion would occur. Figure 13 shows corrosion at the bottom of a pipeline: Here only the base metal corroded while weld metal and HAZ remained unblemished. (Note the corrosion actually "jumped" the weld). Galvanic corrosion is relevant to Indian Point, as discussed below.

e) **Microbiologically Influenced Corrosion** (MIC) and deposits are important to consider as well, and are detailed in the Brookhaven report[20] but not addressed in the LRA.

B. Factors Affecting Internal Corrosion

1. Corrosion Rates

27. In the absence of detailed failure analyses and root cause identification for the failures observed at Indian Point[21], a few remarks are in order to delineate what can be expected in the future, to estimate the rate of corrosion likely to persist in these lines and discuss the necessary monitoring procedures and the required frequency thereof.

28. At Indian Point Units 2 and 3, most of the buried pipes are internally cement-coated.[22] The type of water carried in the service water system is to a large extent brackish, *i.e.* it originates from the river at a depth of about 10 feet. This level can vary from about 4 feet to as much as 15 feet.[23] As a consequence, the oxygen content in the feed water to the service water system may vary from fully

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[20] The Brookhaven Report, above note 11, at 26-28.

[21] *See* Entergy, Response to Request for Additional Information, above note 14.

[22] *Id.* at Attachment 2, p.1

[23] IP3 FSAR Update, 9.6.1 Service Water Systems, p. 108 of 176.

aerated (about 8 ppm depending on temperature) to partially aerated (about 4 ppm depending on temperature). The level of the oxygen concentration is crucial in estimating the prevailing or future corrosion rates.

29. In view of the fact that the water is brackish, hence contains chlorides, the intake water is very corrosive, to possibly destructive, depending on conditions. Entergy reports for the resistivity of the intake water a range of from 59 ohm-cm to 10,000 ohm-cm.[24] These values, according to Figure 5 (attached), would correspond to a chloride content of the intake water of from 7,000 ppm to as little as 30 ppm.[25] (Corrosion rates are most often related to chloride concentration rather than conductivity or the inverse, resistivity.) The average resistivity of the intake water is reported as 300 ohm-cm or about 1,000 ppm chlorides. It is not clear whether this is a time-weighted average or a straight mean (if it were time-weighted, the chloride content would likely be different than estimated). It is difficult to determine the precise chloride level of the intake water – most of the time it is of the order of between 5,000 to 10,000 ppm, but it can be as low as 2,000 to 5,000 ppm (see Figure 5-A, attached) depending on river flow (sea water is 19,000 ppm), with the water being near fully oxygenated. The temperature is ambient most of the time (50 to 75^0 F). Estimation of corrosion rates under these condition is not straightforward and must take into account different sources in order to arrive at some bracketed ranges. For instance, the damage shown in Figure 4 (attached) for a 6-inch internal diameter cast iron elbow occurred in about 4 months with fully aerated salt water at room temperature, but also at high velocities (30 to

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[24] IP2 FSAR Update, 5.1.3.12 (Chapter 5, p. 38 of 89, Revision 20 (2006)).

[25] See for instance A.G. Ostroff, Introduction to Oilfield Water Technology, NACE, 1979, p. 381. The chart at Figure 5 was created by referencing the table within this document that correlates chloride to resistivity.

40/ft/sec).[26] The estimated maximum local corrosion rate, which led to perforation in 4 months, was of the order of 500 to 1,000 mils per year (mpy) or 0.5 to 1 inch per year (ipy).

30. On the other hand, as shown in Figure 6 (attached), the corrosion rate in mildly stirred tap water (~100 pm chloride) at ambient temperature was measured at between 70 and 100 mpy.[27] But this result applies to a short-term measurement when no protective corrosion product layer (rust) has had time to form yet, as it likely has by now on pipes at Indian Point Units 2 and 3. However, it is precisely these corrosion rates that apply to high velocity situations where corrosion product layers are washed away due to high turbulence and high shear stresses. The corrosion literature often reports smaller corrosion rates for steel in aerated low TDS (total dissolved solids) water, between 10 and 20 mpy.[28] Entergy quotes a "typical" corrosion rate of 12 mpy, but does not indicate the range of what it considers "typical."[29] According to LaQue, velocities of the order of 15 to 20 ft/sec will increase these corrosion rates in sea water to about 40 mpy.[30] The point of this discussion is to demonstrate that ("general") corrosion rates[31] can vary within large limits (as also pointed out by the Brookhaven Report, pg. 33). It is possible that

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[26] Photograph by R.H. Hausler, circa late 1970s.

[27] R.H. Hausler, laboratory results, electronic resistance probe, Mobile Bay, Alabama (2003).

[28] *See* Figure 7, attached (excerpt from H.H. UHLIG, CORROSION AND CORROSION CONTROL, p. 84, 1963, John Wiley and Sons, New York).

[29] Entergy, Response to Request for Additional Information, above note 14, Attachment 1, p. 1 of 2.

[30] H.H. Uhlig, above note 28, at 96.

[31] As opposed to pitting and localized corrosion affected by metallurgical effects.

corrosion is more aggressive in certain locations at higher corrosion rates than the 12 mpy contemplated by the LRA. To the best of my knowledge, Entergy has provided no calculations or analyses, based on the relevant factors at this site, to support the corrosion rate it considered "typical."

2. Metallurgy

31. One can also observe many localized corrosion phenomena which are based on metallurgical effects. Typical examples which also relate to Indian Point are shown in Figures 8 and 9 (attached).[32] Here one observes selective removal of weld metal from pipeline junctures. Where pipe joints are welded together one can observe, depending on conditions and the prevailing metallurgy that either the weld metal, the heat affected zone (HAZ), or the base metal can corrode preferentially. Many such examples have been observed in studies related to the chemical cleaning of nuclear steam generators.[33] It is therefore dangerous to assert, as Entergy implicitly does, that it has taken ample preventative measures to reasonably assure continued integrity of the pipes even after nearly 40 years of service life when no specific measures are outlined.

32. The failure reported by Entergy in September of 2007 is likely of this nature, involving a weld.[34] "On September 18, 2007 a Nuclear Plant Operator [at Indian Point] conducting a routine plant walkdown noted a minor leakage of approximately 5 drops per minute in one of the two cement-lined 18: diameter 0.375 inch nominal thickness service water supply lines for the containment fan cooler

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[32] Photographs by R. H. Hausler, circa 1984.

[33] R.H. Hausler, Non-proprietary Corrosion Inhibitors for Solvents to Clean Steam Generators, EPRI NP-3030 (June 1983).

[34] Entergy, Response to Request for Additional Information, above note 14.

units."[35] This would indicate to me that the corrosion, which is topographically shown in Figure 3 (attached) occurred either in the weld metal or the HAZ. Importantly, however, it occurred where the cement-coated surface was joined by welding to the bare metal flange. Stated differently, this joinder involved dissimilar materials and a possible crevice (underneath the cement coating) coupled with high velocity aerated saline water. It is somewhat surprising that in light of the above analysis regarding corrosion rates under these condition a penetration rate of only 12 mpy was assessed (or estimated) by Entergy.[36] However, Entergy described this estimate corrosion rate as "typical," meaning the value could have been higher. There are many places at Indian Point where cement lined pipe sections have been joined by welding. Typically the resulting configuration can be represented as is shown in Figure 10 (attached).[37] Where there are welded joints between two sections of pipe there is a breakdown in the cement coating which exposes the weld metal, but also creates a location where flow is upset and becomes highly turbulent, thereby accelerating the corrosion.

33. In view of the fact that Entergy has itself estimated the corrosion rate at 12 mpy (0.012 inches per year)[38] and considering that the plants' age is in excess of 30 years it is now highly likely that every weld in the cement coated service water piping system has reached the end of its useful life span. Therefore, before a license renewal, every weld must be inspected from the inside and the damage assessed to establish a "baseline" analysis for any future Aging Management Plan.

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[35] *Id.* at Attachment 2(D).

[36] Entergy, Response to Request for Additional Information, above note 14.

[37] Diagram based on my understanding of weldments of cement-coated pipes. Other configurations are possible and may have been used, but the corrosion problems would be very much the same.

[38] Entergy, Response to Request for Additional Information, above note 14.

Since it will be virtually impossible to assess the extent of the damage visually from the outside (as the pipes are buried), ultrasonic testing (UT) measurements must be made over the full periphery of each weld from the inside. Special attention must be given to those welds that are located upstream or downstream of a flow disturbance. It will not be possible to assess possible damage below the coating in the pipe body. Therefore, additionally, all piping needs to be pressure tested to at least twice the operating pressure. Inability to perform pressure tests for any reason should not be cause for relief.

34. Because the conditions for corrosion are present at this facility, and because there is evidence that corrosion has already occurred to the point where leaks have been discovered, Indian Point's aging management plan must frequently inspect the welds. Such inspection must in my opinion be based on at least a two-year cycle. Depending on the extent of the damage one may find and the corrosion rates that may result for certain locations from trending on the basis of repeated inspection this inspection cycle may have to be shortened.

C. Factors Affecting External Corrosion

35. While all of these mechanisms listed above (general, pitting, crevice, galvanic, and microbiologically-induced corrosion) apply in some measure to external as well as internal corrosion, deterioration from the outside is also caused by additional phenomena.

36. First, it is basic that water and moisture are needed for external corrosion to occur. Second, safety-related buried pipe (like the service water system) is almost always coated on the outside of the pipe by various means and methods in use at the time of construction, so if corrosion is observed, then by necessity the coating has deteriorated. For this to happen, one again needs water

in most instances, or at least a moist, conductive environment.[39]

37. If moisture is present and the coating has deteriorated one needs additionally a cathodic depolarizer, *i.e.* a substance which furthers the cathodic reaction, in order for corrosion to occur. This may be oxygen as occurs in soil at shallow depth. It could also be certain bacteria or it could be a low pH generated by acid rain. Additionally, underground corrosion is always amplified by stray currents which, in my professional experience, are also always present in one form or another at power generating stations.

38. The UFSAR informs that the circulating water lines are protected by concrete encasement in areas of high corrosion and do not require cathodic protection.[40] However, if a corrosion leak were to occur from the inside of the pipe, then the concrete encasement, or chase, would become wet, which would exacerbate external corrosion. The LRA does not address specific measures taken to guard against the occurrence of this phenomena.

39. The soil was inspected by A.V. Smith Engineering Company some time prior to 1968, and this inspection was apparently detailed in a report which does not appear to be available.[41] Absent this information, there is no way to examine the veracity of the methodology used, or the assumptions or factors considered during the inspection. A decision was made on the basis of this inspection, in the 1960s, not to cathodically protect the reactor building liner, nor the buried pipes.

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[39] By "conductive," as applied to soil or the environment around a buried pipe, one understands electrolytic conductivity (as opposed to electrical conductivity as is inherent in metals).

[40] UFSAR, Indian Point Unit 3, p.59.

[41] Final Facility Description and Safety Analysis Report, Consolidated Edison Company of New York, Indian Point Nuclear Generating Unit No. 2 (Oct. 15, 1968).

Nevertheless, protective coating was recommended to eliminate random localized corrosion attack.[42] It does not appear that cathodic protection was installed, but that there was enough random variation in the soil survey to suggest that external corrosion of carbon steel pipe was a real possibility and hence an external coating was required.

40. Sandy clay, such as surrounds Indian Point Units 2 and 3,[43] will retain humidity/moisture for at least some time. It is therefore likely that the soil resistivity numbers quoted by Entergy in the UFSAR as being mostly above 10,000 ohm-cm are too high. It is therefore entirely reasonable to expect that many of the assumptions made 35 years ago with respect to the need for cathodic protection (*i.e.*, those based on soil conditions and moisture levels which may be different today than a the time of the A.V. Smith analysis) are no longer adequate.

41. If in fact some of the pipe had been coated to prevent random corrosion, as had apparently been anticipated by A.V. Smith, it is entirely possible that some of the coating may have deteriorated. The GALL report states that "corrosion pits from the outside diameter have been discovered in buried coating piping in far less than 60 years of operation."[44]

42. According to the National Association of Corrosion Engineers (NACE), which establishes industry standards applicable to the all buried piping systems including these, the assessment of the condition of the coating and cathodic

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[42] USFAR, Indian Point Unit 3, § 16.4.4 (Cathodic Protection).

[43] UFSAR, Indian Point Unit 2, § 5.1.3.12; USFAR, Indian Point Unit 3, §16.4.4.

[44] United States Nuclear Regulatory Commission, Generic Aging Lessons Learned (GALL) Report, NUREG 1801 (July 2001)(hereinafter the "GALL Report").

protection system (if any) is to be conducted on an annual basis and compared to predetermined values.[45] Entergy has scheduled, as best as I can tell, similar inspections on the10-year cycle. Because of the uncertainty of the current condition of these coated (but non-cathodically-protected pipes), Entergy's Aging Management Plan should follow the NACE standards for coated pipe inspections.

V. Requirements for Preventative Maintenance and Predictive Monitoring

43. According to the LRA Technical Information Appendix B "Aging Management Programs and Activities" (AMP) section, B.2.6 "Buried Piping and Tanks Inspection," the AMP is a

> new program that includes a) preventive measures to mitigate corrosion and b) inspections to manage the effects of corrosion on the pressure retaining capabilities of buried carbon steel, gray cast iron and stainless steel components. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried components are inspected when excavated during maintenance. If trending within the corrective action program identifies susceptible locations, the areas with a history of corrosion problems are evaluated for the need for additional inspection, alternate coating, or replacement. This program applies to among other systems, to the Service Water System.

44. Section B.1.6 which deals with buried pipe and is the only program I found to deal with these structures, is limited to external corrosion damage. Second, the program references preventative measures; however, as much as I focused to find such mitigation measures in the LRA or Appendix B, I cannot find any preventative measures in the LRA which would mitigate external corrosion. One therefore must conclude that the program is essentially credited only with *post*

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[45] *See* National Association of Corrosion Engineers (NACE) Standards RP-0285-95 and RP-0169-96 and their updated versions (SP-0285-2002 and SP-0169-2002).

facto excavation and maintenance, once damage has been detected by leaking fluids to the surface. If no such leaks are detected at the surface the failure may continue to spill radioactive water into the ground until such time that inventory imbalance becomes too obvious.

45. NRC and the industry as a whole have recognized that preventive measures are important and that leaks should not be tolerated (see the report prepared on Aging of Buried Pipes prepared by the Brookhaven National Laboratory and published by NRC,[46] and the GALL report[47]). These reports discuss in detail virtually all the possible mechanisms for the degradation of the integrity of piping and in particular buried piping systems. The GALL report also references recommended practices by the National Association of Corrosion Engineers (NACE) for the prevention of external degradation.[48]

46. The "new program" mentioned in the LRA, Appendix B (pg B-27) appears to be limited to post-accident maintenance, meaning that whenever a pipe has to be excavated because of the detection of a leak, it will be repaired "in accordance with standard industry practice for maintaining external coatings and wrappings." It is not clear whether this inspection extends to possible internal damage as well.

47. The Brookhaven report states that the degradation of buried pipe is a concern that needs to be addressed on an ongoing basis.[49] The NRC has established

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[46] The Brookhaven Report, above note 11.

[47] The GALL Report, above note 44.

[48] *Id.*

[49] The Brookhaven Report, above note 11, at 29.

in this context the basic maintenance rule which "requires the licensee to monitor the performance or conditions of structures, systems and components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that the SSCs will be capable of performing their intended functions."[50] The LRA is deficient in its approach to meet the requirements of this basic NRC rule because of the reasons addressed above. I therefore think a minimal surveillance and monitoring scheme as outlined below must be adopted by the Licensee of Indian Point 2 and Indian Point 3.

A) Monitoring and Inspection of Buried Pipes for Internal Corrosion

48. Since there is at this time no consistent history compiled on the state of deterioration of the buried pipes at Indian Point 2 and Indian Point 3 (as well as Indian Point 1), the Licensee needs to begin to establish, prior to the license renewal, the true state of integrity of the infrastructure of the two plants. The methodologies by means of which this can be done have been listed and discussed exhaustively in the Brookhaven report.[51] Some minimal inspection is listed below:

> a) All safety-related piping, and all radioactive water carrying piping must be pressured tested to at least twice the operating pressure. A reasonable acceptance criterion (such as rate of loss of pressure, $\Delta p/hr$) must be established for all pipes and accepted by the NRC and must become the basis for the new licensing basis. Leaks must be repaired.
>
> b) Since pressure testing can only be identify corrosion failures, but cannot give information about ongoing deterioration, which has not yet reached the failure stage, alternate means must be applied to determine the extent of wall thinning, weld corrosion, and coating defects. Many such means have been described

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[50] 10 C.F.R. § 50.65 (Maintenance Rule).

[51] The Brookhaven Report, above note 11, at 41-44.

exhaustively in the Brookhaven report. They include visual inspection (remotely operated optical recorder), ultrasonic testing (UT) surveys, intelligent pigs, etc. The choice of method will depend on circumstances but must be such as to furnish a complete assessment of the integrity of the pipe.

c) Evaluation of the results must occur on the basis of statistical methodologies and result in reasonable prediction of failure. For instance, I evaluated the data presented in Entergy's October 3, 2007 Response to request for Relief (see footnote 11, *above*, attachment 4, p.4), by extreme value methodology and the result appears in Figure 14. It can be seen that there is a reasonably high probability that the remaining wall thickness is less that 20% of the original. At a corrosion rate of 12 mpy, total penetration at this location would occur in 6 years. Therefore, re-inspection must be scheduled for at least three years after the observation had been made (9/19/2007).

d) Wherever possible, intelligent pigs must be used and the results statistically evaluated.

e) Instead of straight UT methodology perpendicular to the pipe wall, the guided wave technique should/must be applied in order to determine wall thickness loss remote from the spots where the coating has been removed.

f) The inspection frequency should be half the anticipated time of expected failure (as is, in my profession experience, a standard measure).

g) Additional methodologies have been described in the Brookhaven report.

B) Monitoring and Inspection of Buried Pipes for External Corrosion

49. Since direct assessment of the degree of deterioration of the external surfaces of buried pipes is not practical, except where leaks have occurred and excavation is unavoidable (because these buried pipes are coated but not protected), indirect methods must be used. These should include:

> a) Establishment of the degree of water/moisture in contact with every buried pipe.
>
> b) Potential surveys must be conducted to establish that each and every buried pipe is free from so-called "hot spots", *i.e.* where the potential is substantially above -0.85 V vs. $CuSO_4$ (copper sulfate).[52]
>
> c) Additionally, it must be established that there are no stray currents flowing through any of the buried pipes.
>
> d) Flexible internal probes, known as "intelligent pigs" (for example, magnetic flux leakage and guided wave technology) have the capability of distinguishing between internal or external corrosion damage.
>
> e) As discussed about in section IV, external inspections should take place every year in accordance with NACE standards.

VI) Summary

50. Deterioration of buried pipes in nuclear power plants is now well established. It is axiomatic that failure rate increases over time.[53] As nuclear power plants continue to operate beyond 40 years it becomes essential to assess the effects

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[52] *See also* NACE Standards RP-0169-2002 6.2.2.1.1, Criteria and Other Considerations for Cathodic Protection, at 13.

[53] W. KENT MUHLBAUER, PIPELINE PROTECTION MANAGEMENT MANUAL, 1-6, fig. 1.1 (Common Failure Rate Curve)(3d Ed. 2004).

of age-related degradation of their plant structures, systems, and components (SSCs).[54] I reviewed the License Renewal Applications for Indian Point 2 and Indian Point 3 with a focus on the maintenance/deterioration of buried pipes of essential safety systems.

51. I found that, as specified in LRA documents with their pertinent Appendices, precious little emphasis is put on the state of integrity, continued monitoring, and the concomitant maintenance programs involving buried SSCs.

52. It is therefore in my opinion imperative that the NRC require the Licensee of Indian Point 2 and Indian Point 3, as part of the Aging Management Plan, to begin to establish systematically, prior to a new license being issued, the sate of integrity of the buried SSCs and develop a consistent plan and commitments for monitoring, maintaining, and/or replacing defective structures, because without knowing existing conditions Entergy cannot know the frequency or type of inspections required to adequately address aging management. To the extent the facilities rely on the piping of Indian Point 1, my opinion would extend to that facility.

53. It has been established that weld corrosion has taken place. All welds need to be inspected. Re-inspection on a 10-year cycle, as proposed by the Licensee, at this stage, is totally inadequate. Re-inspection has to be based on the degree of corrosion that has already taken place and the estimated extreme (from extreme value statistics, Figure 14, attached) corrosion rate. The Licensee has estimated a typical corrosion rate of 12 mpy.[55] The Brookhaven report lists a survey of the

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[54] The Brookhaven Report, above note 11, at 1.

[55] Entergy, Response to Request for Additional Information, above note 14 at Attachment 1, p.1 of 2.

various corrosion rates one can expect and cities corrosion rate of from 1 mpy to 100 mpy. This should indicate a 10 year cycle is by no means adequate where no one could assume that on average perhaps as much as 50% of wall thickness (150 mils) has been lost locally, and in extreme cases in excess of 80%. Inspection frequency should not be based on end of life expectation, but should occur at a minimum at half of the time it would take to the occurrence of a failure.

54. Wherever possible, intelligent pigs should be used because with their methodology corrosion both from the inside and outside can be detected. Alternate methodologies such as guided wave technology have been listed above and can be found in the Brookhaven report, and should be used.

55. It is my judgment, after having reviewed a large number of documents relating to the license renewal and isolated failure reports that the NRC should not accept the Licensee's proposed aging management process because of the many shortcomings spelled out above.

56. Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Rudolf H. Hausler

Rudolf H. Hausler

Executed this 26 day of November, 2007

Figures Attached to the Declaration of Rudolf H. Hausler

Figure 1

Emergency Service Water Pipe Emerging from Underground

Figure 2

Same as above illustrating the large amount of rust on the Outside of the pipe.

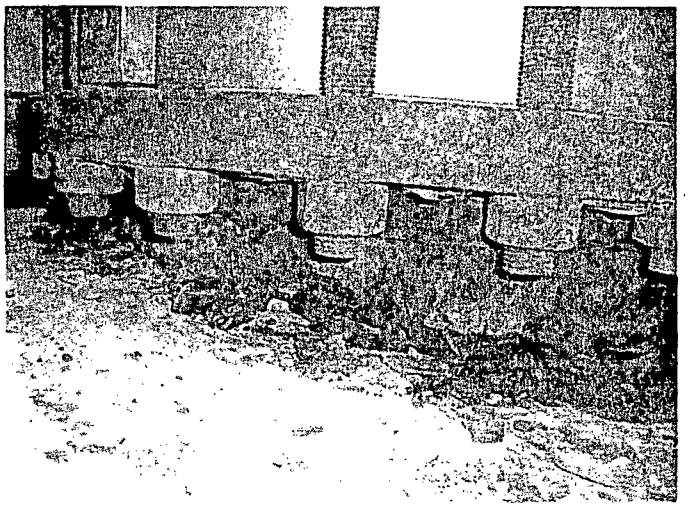


Figure 3

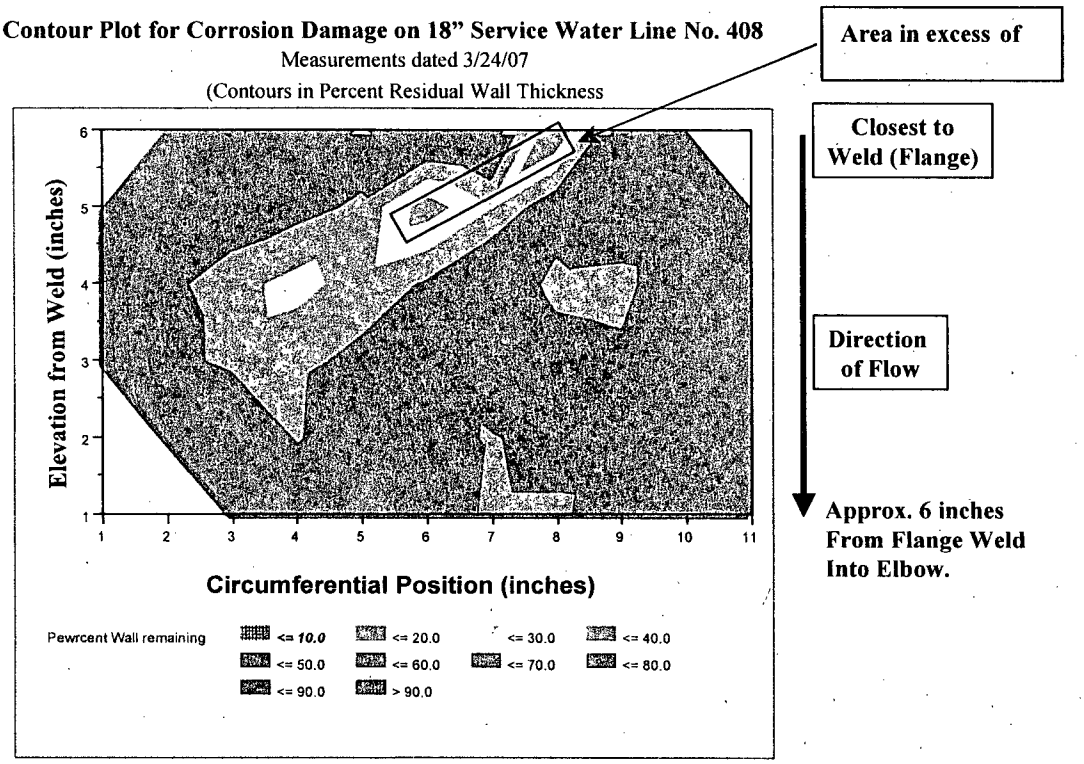


Figure 4

Flow induced Localized Corrosion due to Aerated Brackish Water

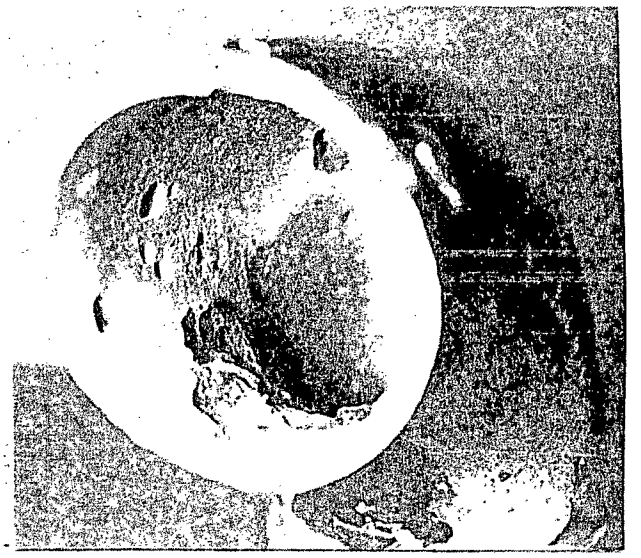


Figure 5

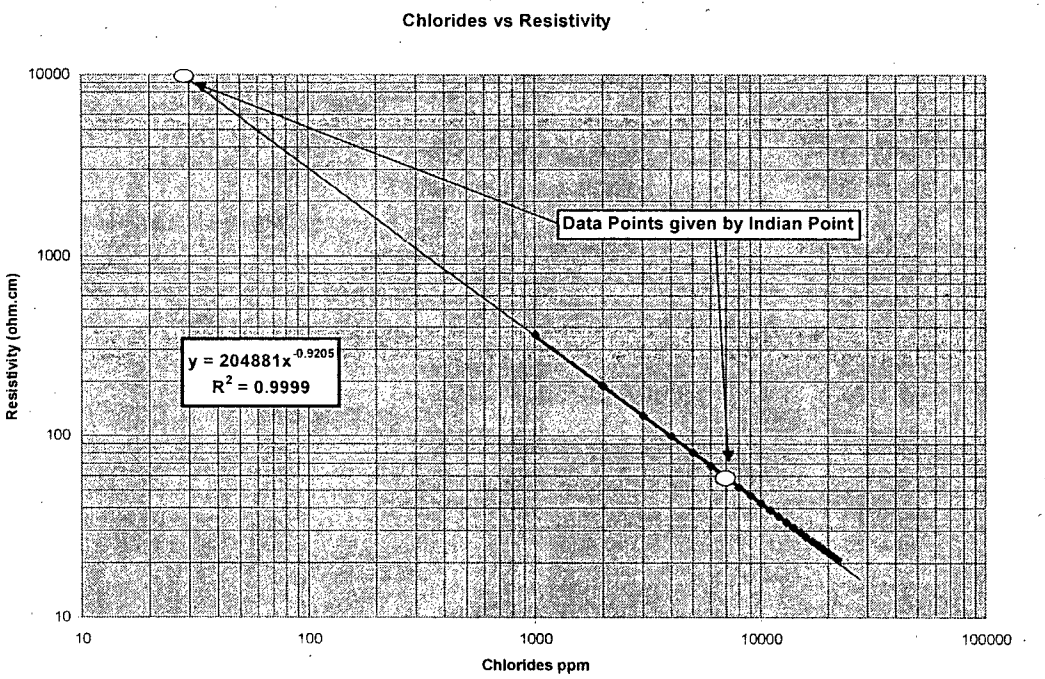
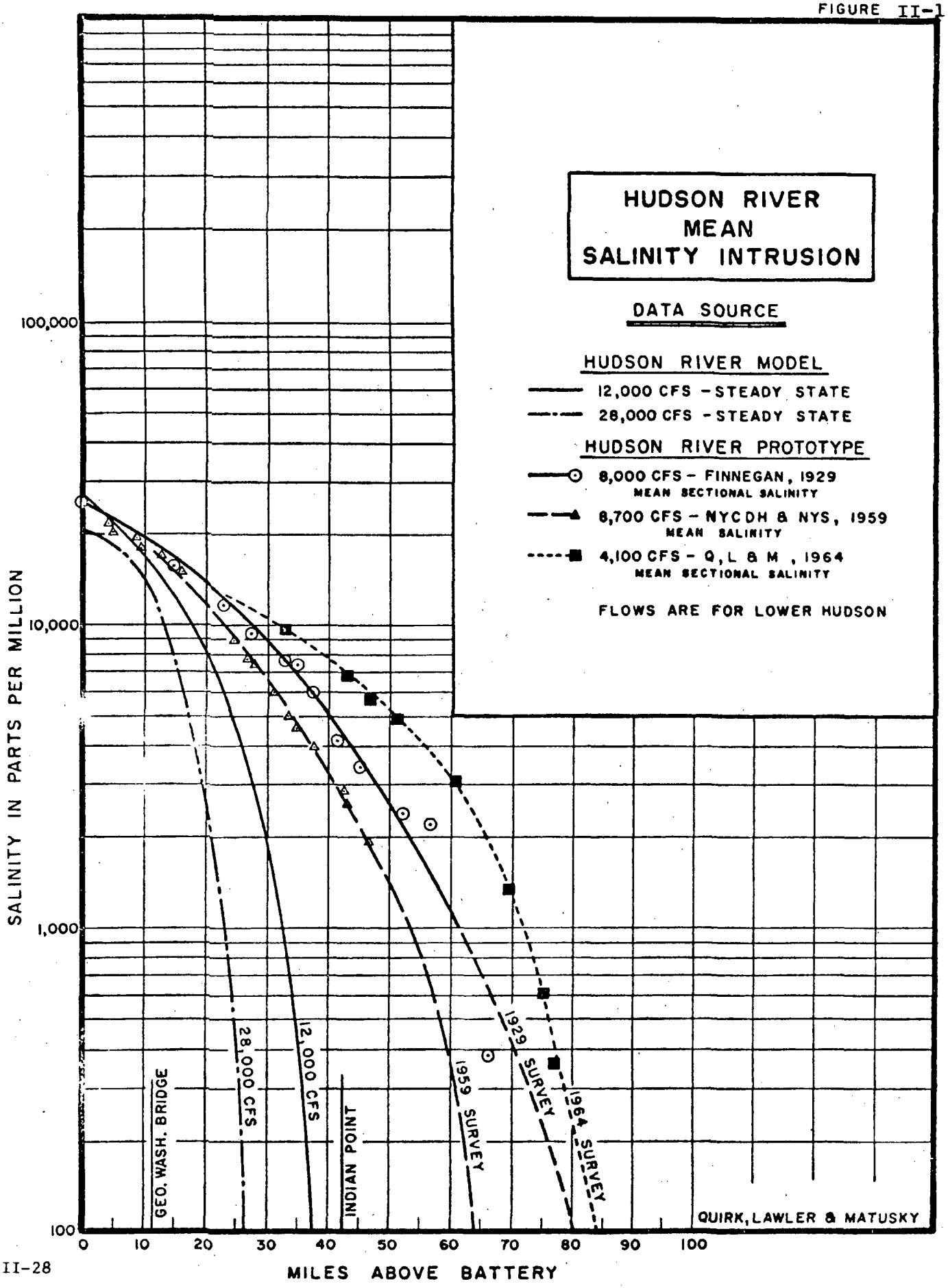

Chlorides vs Resistivity
10000
1000
100
10
Resistivity (ohm.cm)
10
100
1000
10000
100000
Chlorides ppm
Data Points given by Indian Point
y = 204881x^-0.9205
R^2 = 0.9999

FIGURE II-11


HUDSON RIVER
MEAN
SALINITY INTRUSION
DATA SOURCE
HUDSON RIVER MODEL
12,000 CFS - STEADY STATE
28,000 CFS - STEADY STATE
HUDSON RIVER PROTOTYPE
8,000 CFS - FINNEGAN, 1929
MEAN SECTIONAL SALINITY
8,700 CFS - NYCDH & NYS, 1959
MEAN SALINITY
4,100 CFS - Q, L & M, 1964
MEAN SECTIONAL SALINITY
FLOWS ARE FOR LOWER HUDSON
SALINITY IN PARTS PER MILLION
100,000
10,000
1,000
100
0
10
20
30
40
50
60
70
80
90
100
MILES ABOVE BATTERY
GEO. WASH. BRIDGE
INDIAN POINT
28,000 CFS
12,000 CFS
1959 SURVEY
1929 SURVEY
1964 SURVEY
QUIRK, LAWLER & MATUSKY

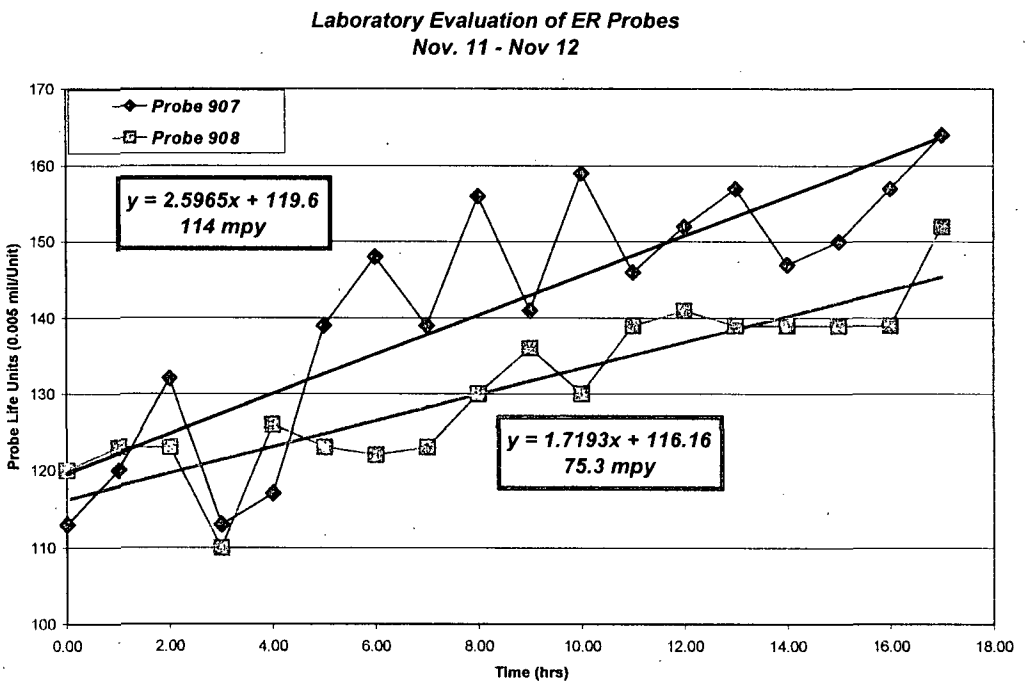

Laboratory Evaluation of ER Probes
Nov. 11 - Nov 12
Probe 907
Probe 908
y = 2.5965x + 119.6
114 mpy
y = 1.7193x + 116.16
75.3 mpy
Probe Life Units (0.005 mil/Unit)
Time (hrs)
170
160
150
140
130
120
110
100
0.00
2.00
4.00
6.00
8.00
10.00
12.00
14.00
16.00
18.00

Figure 6

Figure 7

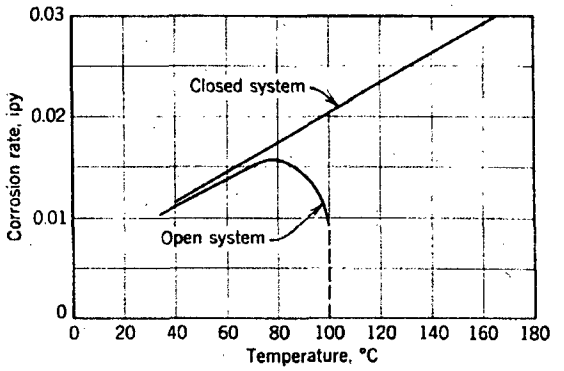


Fig. 3. Effect of temperature on corrosion of iron in water containing dissolved oxygen (*Corrosion, Causes and Prevention*, F. Speller, p. 168, McGraw-Hill, 1951) (with permission).

Figure 8
Typical Localized Weld Metal Corrosion

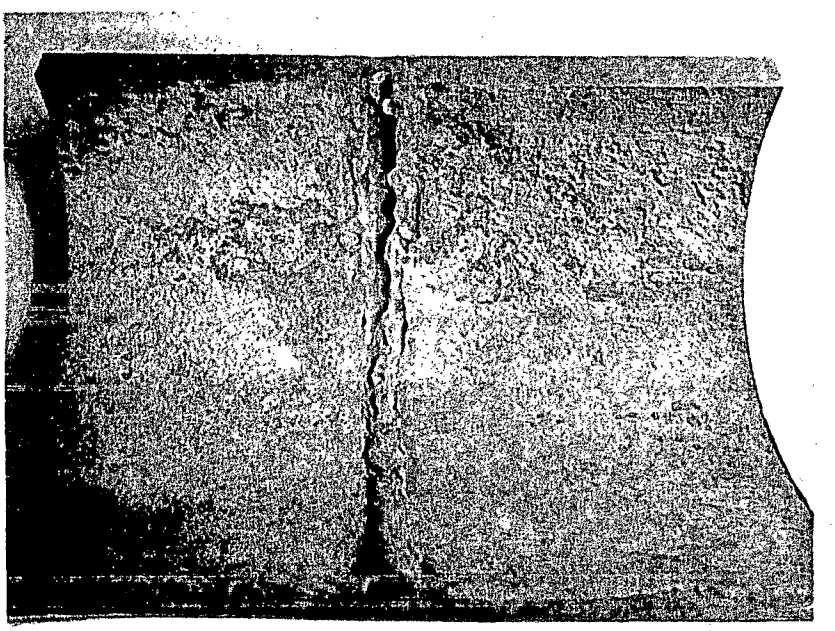


Figure 9

Typical Selective Weld Metal Corrosion

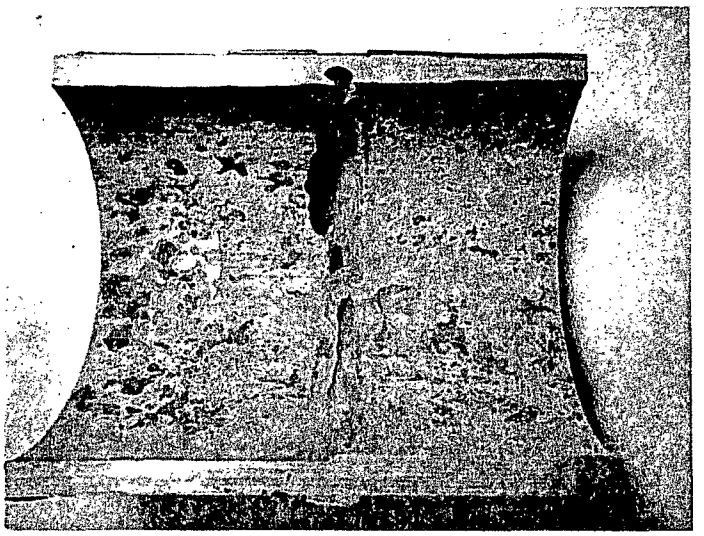


Figure 10

Typical Weld on Internally Cement Coated Pipe With External Coating Wrap

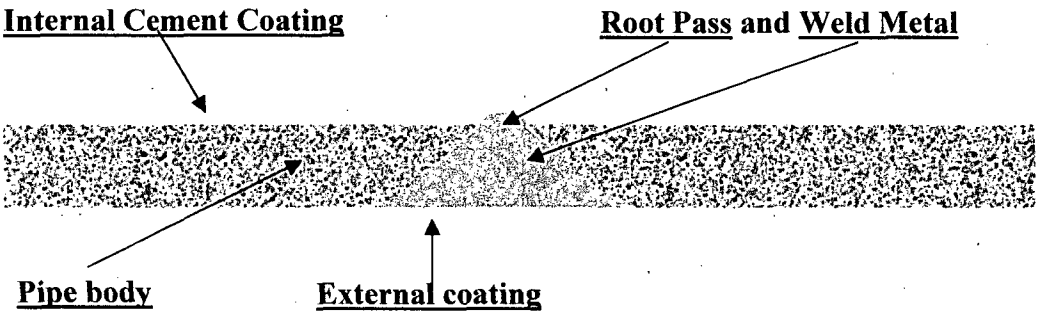


Figure 11
Typical Crevice Corrosion

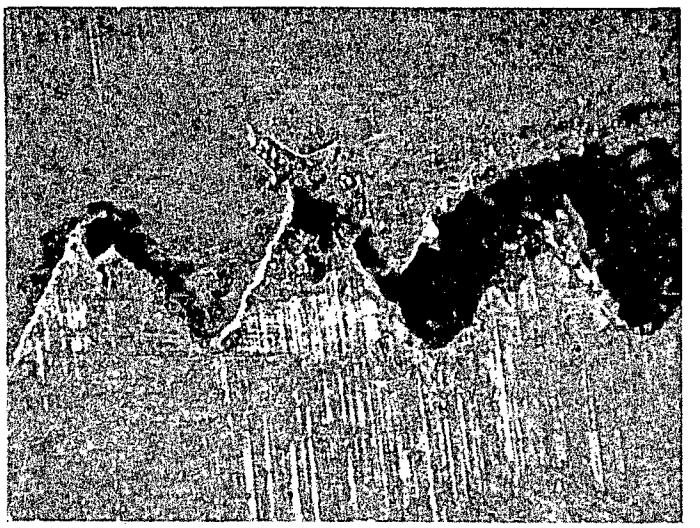


Figure 12
Example of Flow Induced Localized Corrosion
(Damage occurred selectively and locally in the HAZ)

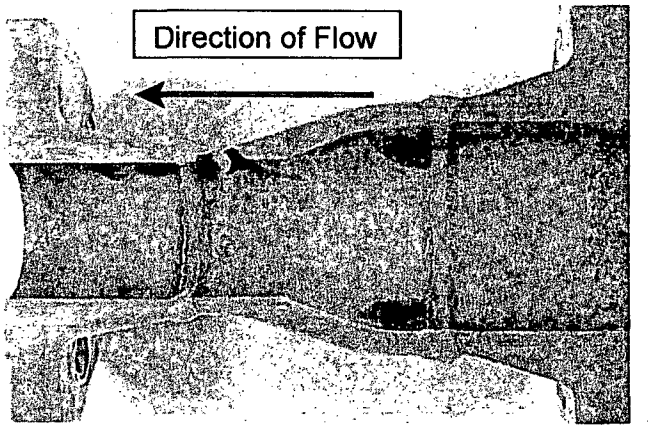


Figure 13

Corrosion on Bottom of Pipeline
(Only Base Metal is Corroded)

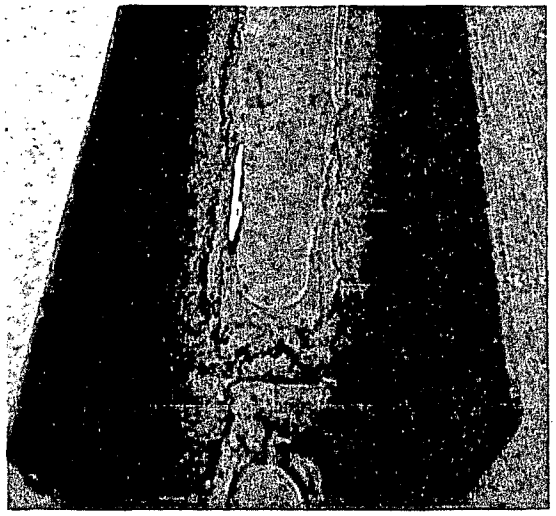
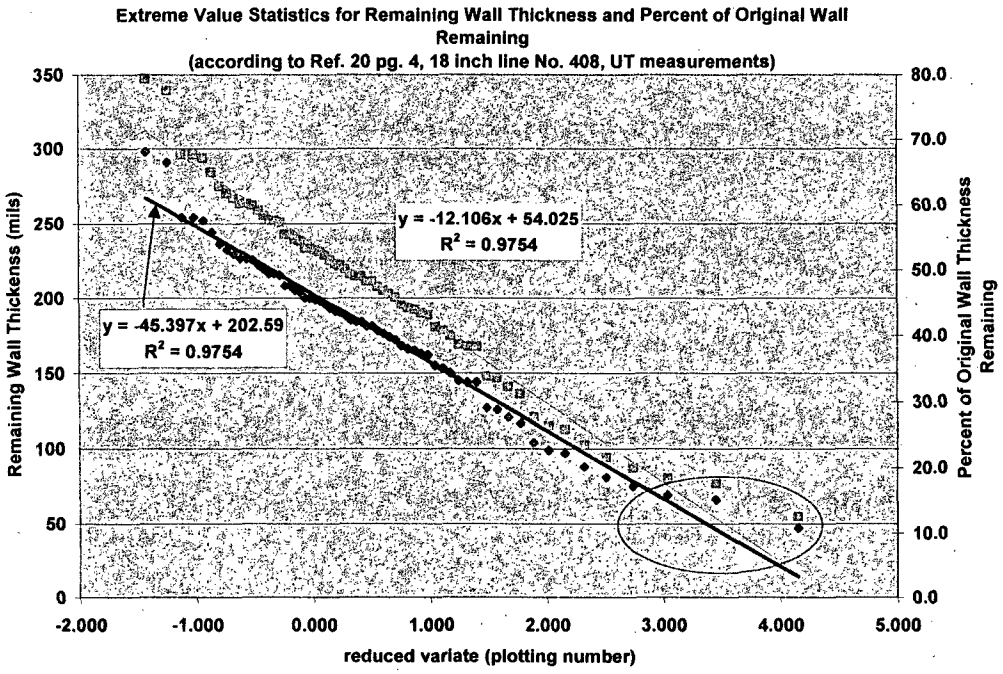


Figure 14



Rudolf H. Hausler

SUMMARY

Over 30^+ years planned, conducted, and directed advanced chemical research focused on oil production and processing additives. Acquired expertise in corrosion prevention, chemical inhibition, and materials selection, failure analysis, trouble shooting and economic analysis. Proficient in German, French, and Italian.

EXPERIENCE:

1996 - Present **CORRO-CONSULTA** (Dallas TX, and Kaufman TX)

President private Consulting Company

Consulted with major Oil Companies on selection, testing and application of Oil Field Chemicals, primarily corrosion inhibitors.

- Worked on Global Sourcing Team for Mobil Oil Company (major fulltime 6^+ months study)
- Consulted for Mobil Oil Company on production chemical usage at Mobile Bay sour gas production field and prepared for changeover to alternate chemical supplier (two year project).
- Consulted for Arco Oil company
    - on sour production in Middle East
    - reviewed North Slope corrosion data (statistical evaluation)
- Consulted for Mobil Oil Company at major CO_2 flood in Oklahoma (extensive laboratory and field testing - two major publications)
- Consulted with Teikoku Oil Company (Japanese National Oil Company) on various subjects of
    - drill string corrosion
    - amine unit corrosion of 304 stainless steel
    - corrosion of 13%-Cr in sweet production and the chemical inhibition thereof
    - identifying qualified corrosion testing laboratories in the US and the world
    - application limits for 3% Cr-steels in oil and gas production
- Consulted for Exxon Mobil on new sourcing study for combined Mobile Bay operations. (Developed novel approach for bid procedure and evaluation of bids on purely technical basis. Developed long-range approach to streamlining operations with potentially large savings.)
- Consulting for Oxy Permian Ltd. on major gas gathering system (changing from dry gas gathering to wet gas gathering)
- Prepared several major publications (see list of publications)
- Major consulting contract for ExxonMobil in Indonesia
- Consulting with various smaller Producers in the US (incl. Anadarko Petroleum Corp and Swift Energy Company)

- Consulting with various engineering companies (e.g. Stress Engineering Services Inc.)
- Consultant on call for Blade Energy Partners
- Consulted with various organization concerned with nuclear safety, including the safety of spent fuel storage casks.

1991 - 1995 **MOBIL Oil Company** (Dallas Research Center), Dallas, Texas

Senior Engineering Advisor

Developed corrosion testing facilities for basic research and to meet specific oil field requirements.

- Planned and developed H_2S corrosion test facility
- Planned safety and wrote safety manual
- Developed unique continuous flow-through corrosion test facility ($$ 1.5MM)
- Developed test protocols and supervised operations of the FTTF
- Extensive consultation with Affiliates on problem solving and chemical usage
- Established supplier relationships and consulted with Affiliates on establishing Enhanced Supplier Relationships
- Developed theory and practice of novel approach to autoclave testing

1979 - 1991 **PETROLITE CORPORATION** St. Louis, Missouri

Research Associate 1986 - 1991

Directed and conducted the development of novel corrosion inhibitors for extreme operating conditions

- New corrosion inhibitor to combat erosion corrosion of carbon steel in gas condensate wells
- Extensive studies on CO_2 corrosion aimed at establishing predictive corrosion model
- Developed the only qualified corrosion inhibitor for nuclear steam generator cleaning (EPRI publication NP-3030 June 1983)

Special Assistant to Executive Vice President 1985 - 1987

Special Assignments focused at support of International Sales

- Extensive travel to secure major accounts in Europe, Russia and East Asia
- Monitored out-sourced R&D in Germany and England

Senior Research Scientist 1979 - 1985

- Developed novel chemical composition under contract with EPRI for corrosion inhibition of cleaning fluids used in nuclear steam generators and methodology of application (only effective formulation still used today)

- Developed unique corrosion model for CO_2 corrosion in oil and gas wells
- Conducted numerous detailed field studies to establish case histories of chemical performance and applications technology

1976 - 1979 **Gordon Lab, Inc.,** Great Bend, Kansas

Technical Director

Responsible for all technical issues involving formulation, application and sales of sucker well production chemicals (corrosion, emulsion, scale, bacteria)

- Conducted failure analysis for customers and developed pertinent reports
- Supervised service laboratory
- Established technical training of sales and support personnel
- Developed technical sales literature and company brochure

1963 - 1976 **UOP (a division of SIGNAL COMPANIES)** Des Plaines, Illinois

Research Associate	1972 - 1976
Associate Research Coordinator	1967 - 1972
Research Chemist	1963 - 1967

To conduct research in electrochemistry, analytical methods development, heat exchanger fouling processes and refinery process additives

- Developed novel organic electrochemical synthesis procedure
- Developed unique (patented) test apparatus for measuring anti-foulant activity
- Introduced statistical design and evaluation of experiments to R&D department and Developed 20 hr course on statistics.
- Developed full 3 credit hour corrosion course to be taught at IIT and DeSoto Chemical Company

EDUCATION

- Ph.D. Chemical Engineering; Swiss Federal Institute of Technology, Zurich Switzerland
- BS, MS Chemical Process Technology, same as above

PROFESSIONAL ASSOCIATION

- American Chemical Society
- The Electrochemical Society
- Society of Petroleum Engineers
- NACE International (Corrosion Engineers)
- American Society fro Metals (ASM)
- Active in NACE on local, regional and national level

RECOGNITION

- NACE Technical Achievement Award (1990)

- NACE Fellow Award 2003

ACHIEVEMENTS
- 17 patents, 58 publications and more than 100 technical presentations
- Registered Professional Engineer (Corrosion Branch, California)
- NACE certified Corrosion Specialist

CORRO-CONSULTA

Rudolf H. Hausler
8081 Diane Drive
Kaufman TX

- **Publications**
- **Books**
- **Awards and Recognition**
- **Patents**
- **Papers Presented before Technical Meetings**
- **Educational Lectures**
- **Continued Professional Education**
- **Professional Activities**

PUBLICATIONS

1. **ELECTROCHEMICAL PRINCIPLES IN CORROSION**
   R. H. Hausler, Heating, Piping, Air Conditioning, 41. (#9) 11(1969)

2. **THE USE OF STATISTICAL DESIGN AND ANALYSIS IN THE DEVELOPMENT OF A CORROSION INHIBITOR TEST**
   R. H. Hausler, L. A. Goeller, R. H. Rosenwald, Proceedings of the National Association of Corrosion Engineers 26th National Conference, March 2-6, 1970, paper #63.

3. **CONTRIBUTION TO THE FILMING AMINE THEORY**
   R. H. Hausler, L. A. Goeller, R. P. Zimmermann, R. H. Rosenwald, Corrosion, 28. (#1) 7 (1972).

4. **CONTRIBUTION TO THE MECHANISM OF HYDROGENSULFIDE CORROSION INHIBITION.**
   R. H. Hausler, L. A. Goeller, R. H. Rosenwald, Proceedings of the 3rd European Symposium on Corrosion Inhibition, Ferrara, Italy, 1971, p. 399.

5. **CORROSION CONTROL IN CRUDE UNIT OVERHEAD SYSTEMS**
   R. H. Hausler, N. Coble, Proceedings of the API Division of Refining, New York, May 8-11, 1972. Also published in: Oil and Gas Journal, 70, (#29) 92, 1972; Hydrocarbon Processing, May 1972, p. 108; Petroliere D'Italia, 19, (#8) 21, 1972; Petroleum Times, June 2, 1972, p. 10.

6. **RUST INHIBITION AND INHIBITOR TESTING, A CRITICAL DISCUSSION**
   R. H. Hausler, R. C. Kunzelman, Materials Protection and Performance, 11, (#11) 27 (1972).

7. **PROCESS CORROSION AND CORROSION INHIBITORS IN THE PETROLEUM INDUSTRY**
   R. H. Hausler, C. Stanski, A. Nevins, Proceedings of the National Meeting of the National Association of Corrosion Engineers, March 1974, Paper #123. Materials Performance, 13, 9 (1974).

8. **FOULING STUDIES IN HYDROCARBON STREAMS IN THE PETROLEUM INDUSTRY**
   R. H. Hausler, Oil and Gas Journal, 71, (#23) 56, (1973).

9. **FREE ENERGY AND ENERGY FUNCTIONS**
R. H. Hausler, Encyclopedia of Chemistry, (Hampil-Hawley, Editors) 3rd Edition, p. 467 (1973).

10. **SYSTEMS APPROACH TO CORROSION ENGINEERING IN THE CHEMICAL INDUSTRY**
R. H. Hausler, Invited paper presented at the AIChE Meeting, Chemical Engineering/Corrosion Interface, Washington, Dec. 1974.

11. **FOULING AND CORROSION IN FEED EFFLUENT EXCHANGER**
R. H. Hausler, C. E. Thalmayer, Discussion of a new Test Method; Proceedings of the API Division of Refining 40th Midyear Meeting, May 13, 1975, Chicago, Ill.

12. **SOME CONSIDERATIONS REGARDING THE USE OF ELECTROCHEMICAL CORROSION MEASUREMENTS IN THE PRESENCE OF CORROSION PRODUCT LAYERS**
R. H. Hausler, Presented at the NACE 1975 North Central Regional Conference, October 21-23, 1975.

13. **INSTRUMENT MEASURES PROCESS FOULING, ETC.**
R. H. Hausler, Oil & Gas Journal, Feb. 14, 1975, p. 92.

14. **CONTRIBUTION TO THE UNDERSTANDING OF FOULING PHENOMENA IN THE PETROLEUM INDUSTRY 16TH NATIONAL HEAT TRANSFER CONFERENCE**
R. Braun, R. H. Hausler, St. Louis, August 11, 1976, Paper #76-CSME/CSChE - 23.

15. **PRACTICAL EXPERIENCES WITH LINEAR POLARIZATION MEASUREMENTS**
R. H. Hausler, Corrosion, 33. (No. 4) 117-1977

16. **CORROSION INHIBITION AND INHIBITORS**
R. H. Hausler, Published in ACS Symposium Series Vol. 89, "Corrosion Chemistry"; p. 262-320, 1977, Jan. 1979.

17. **ECONOMICS OF CORROSION CONTROL**
R. H. Hausler, Materials Performance 17, (#6) 9, 1978

18. **OIL TREATING CHEMICALS - EMULSIONS**
R. H. Hausler, Oil & Gas Journal, September 4, 1978.

19. **SCALING AND SCALE CONTROL IN OILFIELD BRINES**
R. H. Hausler, Oil & Gas Journal, Sept. 18, 1978.

20. **CORROSION INHIBITION AND GALVANIC COUPLES IN THE OILFIELD**
R. H. Hausler, Paper presented NACE: CORROSION/79, Atlanta, Georgia, 1979, paper #17

21. **PREVENTING SCALE AND EMULSION PROBLEMS IN OIL PRODUCTION**
R. H. Hausler, Invited paper presented before the International Petroleum Exhibition, Tulsa, Oklahoma, Sept. 1979

22. **CORROSION MONITORING IN CHEMICAL CLEANING SOLUTIONS**
R. H. Hausler, J. M. Jevec, W. S. Leedy; NACE: Corrosion/83 1983 Paper No. 227 p. 13.

23. **LABORATORY INVESTIGATIONS OF THE CORROSION MECHANISM AS APPLIED TO HOT DEEP GAS WELLS**
R. H. Hausler; NACE: Corrosion/83 1983 paper No. 47, 16 p; Advances in CO_2 Corrosion, Vol. 1, p. 72, (1984).

24. **THE COPRA CORRELATION - A QUANTITATIVE ASSESSMENT OF DEEP HOT GAS WELL CORROSION AND ITS CONTROL**
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25. **THE ROLE OF HYDRAZINE IN EDTA CONTAINING CHEMICAL CLEANING SOLVENTS**
R. H. Hausler; NACE: Corrosion/82 1982 Paper No. 30, 12 p.

26. **CORROSIVENESS OF EDTA CHEMICAL CLEANING SOLUTION**
R. H. Hausler, A. L. Savage; NACE: Corrosion/82 1982 Paper No. 31, 15 p.

27. **INHIBITION OF CORROSION REACTION BY CORROSION PRODUCT LAYERS OF TYPE METAL ION CHELATING AGENTS**
R. H. Hausler; NACE: Corrosion/81, 1981 Paper No. 252, 17 p.

28. **CORROSION INHIBITION AND GALVANIC COUPLES IN THE OILFIELD**
R. H. Hausler, NACE: Corrosion/79 1979 Paper No. 17

29. **NONPROPRIETARY CORROSION INHIBITORS FOR SOLVENTS TO CLEAN STEAM GENERATORS**
R. H. Hausler, EPRI Final Report (346 pp) #NP-3030), Project S-148, June 1983.

30. **THE IMPORTANCE OF THE INTERPHASE IN THE CONTROL OF CORROSION IN EDTA SOLUTIONS**

R. H. Hausler, Presented before the International Conference on Corrosion nhibition, NACE May 1983, published in CORROSION INHIBITION, NACE-7, p.132(1988).

31. **ASSESSMENT OF CO_2 CORROSION IN THE COTTON VALLEY LIMESTONE TREND**

P. A. Burke, R. H. Hausler, presented at NACE Corrosion/84, 1984, paper #288 published in Materials Performance 24 (Aug.) p. 26 (1985).

32. **CORROSION AND CORROSION INHIBITION IN THE COTTON VALLEY PRODUCTION-FIELD EXPERIENCES WITH IRON-COUNTS, COUPONS AND CALIPERS**

R. H. Hausler, P. A. Burke, presented at NACE Corrosion/85, 1985 paper #26 (published in Advances of CO_2 Corrosion Vol. 2, p. 161, 1985).

33. **CORROSION INHIBITION IN THE PRESENCE OF CORROSION PRODUCT LAYERS**

R. H. Hausler, invited paper presented before the 6th European Conference on Corrosion Inhibition, Ferrara, Italy, September 1985, Proceedings p. 41, 1985.

34. **LOW-VOLUME CONTINUOUS INHIBITION TECHNIQUE, AT NOMINAL EXPENSE, FOR HIGH WELLBORE TEMPERATURE, CO_2 AND MILDLY SOUR GAS PROCEDURES**

S. G. Weeks, R. H. Hausler, SPE, Las Vegas, Sept. 1985 paper #14404.

35. **LOW COST, LOW VOLUME, CONTINUOUS INHIBITION OF GAS PRODUCTION TUBULARS**

R. H. Hausler, S. G. Weeks, Materials Performance, June 1986, p. 28 (1986), NACE.

36. **ON THE USE OF LINEAR POLARIZATION MEASUREMENT FOR THE EVALUATION OF CORROSION INHIBITORS IN CONCENTRATED HCL AT 200°F (93°C)**

R. H. Hausler, CORROSION, 42, No. 12, 729 (1986).

37. **CO_2 CORROSION AND ITS PREVENTION BY CHEMICAL INHIBITION IN OIL AND GAS PRODUCTION**

R. H. Hausler, D. W. Stegmann, NACE CORROSION/88, St. Louis, NACE 1988, paper #363.

38. **THE METHODOLOGY OF CORROSION INHIBITOR DEVELOPMENT FOR CO_2 SYSTEMS**

R. H. Hausler, D. W. Stegmann, R. F. Stevens, NACE CORROSION/88 St. Louis, 1988, paper #360, published in Werkstoffe und Korrosion, Febr. 1989, and Corrosion, 45 (10) (1989).

39. **STUDIES RELATING TO THE PREDICTIVENESS OF CORROSION INHIBITOR EVALUATIONS IN LABORATORY AND FIELD ENVIRONMENTS**

R. H. Hausler, D. W. Stegmann, EUROPEC/88, London, SPE paper #18369, 1988, accepted for publication in Petroleum Production Engineering, 1990.

40. **LABORATORY STUDIES ON FLOW INDUCED LOCALIZED CORROSION IN CO_2/H_2S ENVIRONMENTS, I. DEVELOPMENT OF TEST METHODOLOGY**

D. W. Stegmann, R. Hausler, C. I. Cruz, H. Sutanto, presented at 6th Asian Pacific Corrosion Control Conference, Singapore, September 1989; also accepted for CORROSION/90, NACE 1990, paper #90005.

41. **LABORATORY STUDIES ON FLOW INDUCED LOCALIZED CORROSION IN CO_2H_2S ENVIRONMENTS, II. PARAMETRIC STUDIES ON THE EFFECTS OF H_2S, CONDENSATE, METALLURGY AND FLOWRATE.**

R. H. Hausler, D. W. Stegmann, C. I. Cruz, D. Tjandroso, ibid.

42. **LABORATORY STUDIES ON FLOW INDUCED LOCALIZED CORROSION IN CO_2/H_2S ENVIRONMENTS, III. CHEMICAL CORROSION INHIBITION**

R. H. Hausler, D. W. Stegmann, C. I. Cruz, D. Tjandroso, ibid.

43. **THE COPRA CORRELATION REVISITED**

R. H. Hausler, T. D. Garber, accepted for presentation at CORROSION/90, NACE 1990, paper #90045.

44. **CO_2 CORROSION AND ITS INHIBITION UNDER EXTREME SHEAR STRESS**

G. Schmitt, T. Simon, R. H. Hausler, accepted for presentation at CORROSION/90, NACE 1990 paper #90022.

45. **CHEMICAL INHIBITION OF FLOW INDUCED LOCALIZED CORROSION IN CO_2 CONTAINING MEDIA**

R. H. Hausler, D. W. Stegmann, Proceedings of the 7th European Symposium on Corrosion Inhibitors, p. 1247, 1990, Ferrara, Italy, Sept. 1990.

46. **INTERPRETATION OF LINEAR POLARIZATION MEASUREMENTS**
(Discussion Contribution) DECHEMA Monograph, Vol. 101, Electrochemical Corrosion Testing, p. 109-114, 1986.

47. **LABORATORY STUDIES ON FLOW INDUCED LOCALIZED CORROSION IN CO_2/H_2S ENVIRONMENTS IV: ASSESSMENT OF KINETICS OF CORROSION INHIBITION BY CONTINUOUS HYDROGEN EVOLUTION MEASUREMENTS.**
R.H. Hausler, D.W. Stegman, paper No. 474 NACE CORROSION/91, (also published in Corrosion Magazine in 1992)

48. **METHODOLOGY FOR CHARGING AUTOCLAVES AT HIGH PRESSURES AND TEMPERATURES WITH ACID GASES**
R.H. Hausler, CORROSION, Vol. 54, No. 8, pg. 641, 1998

49. **CO_2 EROSION CORROSION AND ITS INHIBITION UNDER EXTREME SHEAR STRESS, II. PERFORMANCE OF INHIBITORS**
G. Schmitt, T. Simon. R.H. Hausler, CORROSION/93, PAPER No. 86, (1993)

50. **DEVELOPMENT OF A CORROSION INHIBITION MODEL: I. LABORATORY STUDIES**
R.H. Hausler, T.G. Martin, D.W. Stegmann, M.B. Ward, CORROSION/99, paper No. 2, NACE 1999

51. **DEVELOPMENT OF A CORROSION INHIBITION MODEL: II. VERIFICATION OF MODEL BY CONTINUOUS CORROSION RATE MEASUREMENTS WITH NOVEL DOWNHOLE TOOL**
T.G. Martin, M.T. Cox, R.H. Hausler, R.J. Dartez, P. Pratt, J.C. Roberts, CORROSION/99, paper No. 3, NACE 1999.

52. **EVALUATION OF COMMERCIAL CORROSION INHIBITOR FOR CO_2 CORROSION OF API-13CR TUBING**
Y. Tomoe, M. Shimizu, R.H. Hausler, CORROSION/2000, PAPER No. 16, NACE 2000.

53. **SOLVING IRON SULFIDE PROBLEMS IN AN OFFSHORE GAS GATHERING SYSTEM**
J.K. Brownlee, J.A. Dougherty, T. Salma, R.H. Hausler, CORROSION/2000, PAPER No. 104, CORROSION/2000, NACE 2000

54. **CORROSION MANAGEMENT IN THE ARUN FIELD**
L.M. Riekels, R.V. Seetharam, R.M. Krishnamurthy, C.F. Kroen, J.L. Pacheco, R.H. Hausler, V.A.M. Semerad, CORROSION/96, paper No. 24, NACE, 1996

55. **DEVELOPMENT OF A CORROSION INHIBITION MODEL FOR OIL AND GAS PRODUCTION BASED ON LABORATORY AND FIELD MEASUREMENETS**
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56. **HYDRODYNAMIC AND FLOW EFFECTS ON CORROSION INHIBITION**
R. H. Hausler, G. Schmitt, CORROSION/2004, paper 04402, NACE 2004

57. **CONTRIBUTION TO THE UNDERSTANDING OF H_2S CORROSION**
R. H. Hausler, CORROSION/2004, paper 04732, NACE 2004.

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R. H. Hausler, Accepted for Publication in *Corrosion Tests and Standards:* ASTM Series: MNL 20, 2nd Edition, in preparation, Robert Baboian, Editor.

BOOKS

1. **CORROSION IN OIL AND GAS PRODUCTION**
L. E. Newton, R. H. Hausler, NACE, Houston, TX. 1983.

2. **ADVANCES IN CO_2 CORROSION VOL. 1**
R. H. Hausler, H. L. Godard, NACE, July 1985.

3. **CORROSION INHIBITION**
R. H. Hausler, NACE-7, 1988.

4. **FLOW INDUCED CORROSION: FUNDAMENTAL STUDIES AND INDUSTRY EXPERIENCE**
K. J. Kennelley, R. H. Hausler, D. C. Silverman, NACE 1991

AWARDS AND RECOGNITION

1. NACE Technical Achievement Award, May, 1990.

2. Plenary Lecture, 6th Asian Pacific Corrosion Control Conference, Singapore, Sept. 1989.

3. Plenary Lecture, 7th European Symposium on Corrosion Inhibitors, Ferrara Italy, Sept. 1990.

4. Plenary Lecture, 5th Middle East Corrosion Conference, Bahrain, Jan. 1991.

5. Invited to present Plenary Lecture 8th European Symposium on Corrosion Inhibitors, Ferrara , Italy, 1995

6. Elected NACE Fellow, May 2003,

PATENTS

USP 3 790 496	R. H. Hausler;	Alkylene Polyamine Polymeric Reaction Product Corrosion Inhibitor
USP 3 609 549	R. H. Hausler, R.W. Sampson;	Corrosion Measuring Device
USP 3 810 009	R. H. Hausler, R.W. Sampson;	Apparatus for Measuring Fouling of a Test Specimen
USP 3 731 187	R. H. Hausler, R.W.Sampson;	Temperature Compensated Fouling Measuring Method and Apparatus
USP 3 705 109	R. H. Hausler, L.A. Goeller;	Corrosion Inhibiting Composition and Use Thereof
USP 3 622 503	R. H. Hausler;	Hydrogen Transfer Agent for Slurry Processing of Hydrocarbonaceous Black Oil
USP 3 562 138	R. H. Hausler;	Structural Element for Use in an Electrolytic Cell
Swiss Patent 4393/62	V. Spreter, R. H. Hausler;	Electrode pour Element Galvanique
USP 3 696 049	R. H. Hausler, L.A. Goeller;	Corrosion Inhibiting Composition and use Thereof
USP 3 696 048	R. H. Hausler, L. A. Goeller;	Corrosion Inhibiting Composition and use Thereof
USP 3881 957	R. H. Hausler;	Electrochemical Cell Comprising a Catalytic Electrode of a Refractory Oxide and a Carbonaceous Pyropolymer

USP 3 913 378	R. H. Hausler;	Apparatus for Measuring Fouling on Metal Surfaces
USP 3 923 606	R. H. Hausler;	Prevention of Corrosion
USP 3 972 732	R. H. Hausler;	Electrochemical Cell
USP 4 454 006	R. H. Hausler, L. Savage, J. B. Harrell;	Method and apparatus for Measuring Total Corrosion Rate
USP 4 495 336	R. H. Hausler, N.E.S. Thompson;	Mercapto-Polycarboxylic Acids
EP 027 5651	R. H. Hausler, B.A. Alink, M. E. Johns, D. W. Stegmann;	Carbondioxide Corrosion Inhibiting Composition and Method of Use Thereof.
EP 927 5646	R. H. Hausler,	Carbon Dioxide Corrosion Inhibiting Composition and Method of Use Thereof.

PAPERS PRESENTED BEFORE TECHNICAL MEETINGS

by R. H. Hausler

1. Corrosion in H_2S Containing Media, before the NACE, T8-2 Committee, January, 1970.

2. Corrosion and Corrosion Inhibition in H_2S and Cl^- Containing Media, before the Montreal, Canada, Section of the NACE, December 8, 1970.

3. Time Effects on Polarization Measurements, before the Chicago Section of the NACE, May 18, 1971.

4. Rust Inhibition and Inhibitor Testing, before the North Central-Northeast NACE Regional Conference, October 16-18, 1972.

5. Process Side Fouling of Heat Exchangers, before the NACE T8-2 Committee, Chicago, March 8, 1973.

6. On the Mechanism of Hydrochloric Acid Inhibition by Organic Molecules; Presented at the Gordon Research Conference on Corrosion, July 1974.

7. On the Mechanism of Corrosion Inhibition by Organic Chemicals before the Chicago Section of the Electrochemical Society, January 9, 1975.

8. Seminar on Corrosion and Fouling in the Petroleum Industry. Full day seminar given before the engineers of the Peruvian Petroleum Company (Petroperu) upon invitation, June 22, 1975, Lima, Peru.

9. Linear Polarization Technique, Paper presented before NACE-Corrosion/76, Houston, Texas, March 22-26, 1976.

10. Corrosion Inhibition. Presented as ACS/NACE/ECS sponsored short course on Chemistry in Corrosion, Chicago, February 24, 1976.

11. Corrosion Inhibitors and Sulfide Corrosion, presented before the NACE Western Kansas Section, Great Bend, KS., May 7, 1976.

12. Cooling Water Treatment, Presented before the first annual Corrosion Control Seminar Sponsored by the Kansas Section of the NACE, November 9, 1977, Great Bend, KS.

13. Economics of Corrosion Control, Dinner Talk before the first annual Corrosion Control Seminar sponsored by the NACE Great Bend, KS. Section.

14. Corrosion Inhibition and Galvanic Couples in the Oilfield, NACE North Central Regional Meeting, October 19-21, (1981).

15. Mechanism of Corrosion Inhibition with Reference to Automotive Coolants, NACE North Central Regional Meeting, October 19-21 (1981).

16. CO_2 Corrosion in the Oil and Gas Production, and Overview, NACE South Central Regional Meeting, Oklahoma City, October, 1983.

17. CO_2 Corrosion in Oil and Gas Production, An Overview, presented before the Corrosion Center of the University of Manchester Institute of Technology, July 4, 1985.

18. New Mechanism for Pitting of Carbon Steel in Inhibited Hydrochloric Acid, presented before the faculty of the Materials Engineering Department of the University of Ferrara, June 24, 1985.

19. Metallurgical Effects on Corrosion Inhibition, presented before the faculty of the Corrosion Center of the Institute for Technical Chemistry and Petroleum Chemistry at the University of Aachen, June 28 (1985).

20. CO_2 Corrosion and Prevention. Formal seminar presentation at NAM-Assen (Holland) June 20 (1985).

21. Systems Approach to Corrosion Engineering as Applied to Oil and Gas Production, presented before the ALL-Union Union Oil Institute, Krasnodar, Russia, June 1985.

22. The Effect of Ohmic Resistance on Linear Polarization Measurements for Corrosion Rate Determination, presented before the NACE Chicago Section, October, 1973.

23. Corrosion Monitoring in Sweet Production, NACE, Canadian Region Western Conference, Calgary, Febr. 25 (1986).

24. Overview of the CO_2 Corrosion Mechanism and Inhibition of Erosion Corrosion, NACE South Central Region Committee, Lafayette, Nov. 16-18, 1987.

25. Novel Approach Toward Assessing Inhibitor Cost for CO_2 Corrosion: Example of a CO_2 Flood, ibid.

26. Systems Approach to Corrosion Inhibition of Gas and Gas Condensate Producing Facilities, Gulf Coast Corrosion Seminar Febr. 1987.

27. Predicting Corrosion Inhibitor Performance - Laboratory Evaluations vs. Field Performance, New Orleans NACE Section Meeting, Jan. 22, 1990.

28. Corrosion Inhibitors in the Oil Field: What do we really put in the Hole? New Orleans Offshore Corrosion Conference, New Orleans, 1996.

29. Corrosion Inhibitors in the Oil Field: What do we really put in the Hole? New Orleans Offshore Corrosion Conference, New Orleans, 1997.

30. Current Status of Corrosion Prediction and Assessment: A Review of Corrosion Modeling. Paper presented before the Midland, TX NACE Section, September, 1996

31. Industrial Corrosion Inhibitors, Lecture presented at Texas A&M on invitation by Professor J. M'O. Bockris, September, 1994

32. Corrosion Inhibition: Quo Vadis; Invited paper presented in German before the Jubilee Symposium for Professor Dr. Gunter Schmitt, Technical University Iserlohn, Germany, September 2002.

33. Failure Prediction and Failure Inhibition in Sour Systems; Discussion Presented before TEG 282X NACE technical committee, April 2003 during Corrosion/2003, San Diego, CA.

34. Pitting Model for H_2S Corrosion; Discussion Contribution presented before the TEG 282X NACE technical corrmittee, March 2004, during CORROSION/2004, New Orleans, LA.

EDUCATIONAL LECTURES

by R. H. Hausler

1. Electrochemistry - a Modern Challenge: presented December 1966 to Science Seminar at Taylor University, Marion, Indiana. February 1970, to Science Seminar of the ACS Student Affiliate Chapter at University of Illinois, Circle Campus, Chicago.

2. Corrosion-5 Billion Dollar Business, presented to an advanced Science Class as Hillcrest High School, Country Club Hills, Illinois, December 13, 1972.

3. Discussion on Cathodic Protection, together with Harry E. Kroon of Illinois Bell Telephone, presented at an Educational Seminar of the Chicago Section, NACE, May 1970.

4. Application of Potentiostatic Techniques in Corrosion Research, presented at an Educational Seminar of the Chicago Section, NACE, October 24, 1970.

5. Electrical Methods for Determining Corrosion Rates, at the 4th Annual Seminar on Fundamentals of Corrosion, Milwaukee School of Engineering, November 23, 1971.

6. Corrosion Prevention in the Chemical Process Industry, both presented at the Summer Engineering Conference on Corrosion Engineering, University of Michigan, Ann Arbor Michigan, June 19-23, 1972, published in the Proceedings.

7. Chemistry of Corrosion, course taught at the Illinois Institute of Technology, Evening Division (Chem 544), 3 credit hours, Jan-May, 1975.

8. Chemistry of Corrosion, course taught for DeSoto, Inc. Research Centre, started Nov. 1975, 15 2-hour lectures.

9. Statistical Design and Evaluation of Experiment, 20 2-hour lectures with examples and applications presented in-house at UOP.

10. Corrosion Engineering, (Course based on MIT Video Tapes), organized 20 seminars at Petrolite and 1/2 hour discussion sessions following review of tapes.

CONTINUED PROFESSIONAL EDUCATION

1. Short Course on Corrosion, University of California Extension, Los Angeles, June 26-30, 1967,.

2. Short Course on Statistical Design and Evaluation of Experiments, University of Detroit, Summer 1966.

3. Engineering Summer Short Course on Statistical Experimental Design, University of Wisconsin Extension, Madison Wisc. June 24-28, 1968.

4. Evolutionary Operations and Non-Linear Estimating, Short Course, Chicago, 1967.

5. R&D and New Venture Management, University of Wisconsin Extension, Madison, Wisconsin, May 15-16, 1969.

6. Industrial Research Institute, Mid-Management Groups Seminar, New York, October 28-30, 1973.

7. Gordon Research Conferences: Electrochemistry 1965, 1966, 1967, 1968; Corrosion 1969, 1971, 1973.

8. Two Phase Gas-Liquid Flow, University of Houston, February 22-26, (1982).

PROFESSIONAL ACTIVITIES

THE ELECTROCHEMICAL SOCIETY

Member since 1964

•	Chicago Section	Secretary	1964-1965
		Treasurer	1966-1967
		Vice-Chairman	1966-1967
		Chairman	1967-1968
		Councilor	1972-1976
•	National Meeting 1968	Co-Treasurer	
•	National Meeting 1973	Hospitality Chairman	

THE NATIONAL ASSOCIATION OF CORROSION ENGINEERS

Member since 1968

•	Chicago Section	Treasurer	1971-1973
		Vice-Chairman	1973-1974
		Chairman	1974-1976
•	National Meeting 1974	General Chairman	
•	North Central Region	Program Chairman	1972
		Regional Meeting	
		Education Chairman	1982-1985
•	Unit Committee T-3A	Chairman	1973-1975
•	Research Committee	Member	1975-1978
		Vice-Chairman	1983-1984
		Liaison Education Committee	1982-1984
		Liaison TPC	1981-1983
•	Education Committee	Member	1983-1985
•	Awards Committee	Member	1984-1986

- Group Committee T-1 Member 1976-
  Vice Chairman
  T-1-3 1981-1984
  Various T-1 Unit Committees and Task Groups

- Organized numerous conferences, among others:

  - International Conference on Corrosion Inhibition, Dallas, 1983

  - International Symposium on CO_2 Corrosion, Los Angeles, 1983

- Co-Editor Advances in Corrosion Inhibition (in preparation).

THE CHICAGO TECHNICAL SOCIETIES COUNCIL

Member 1968

Treasurer	1970-1972
Vice-Chairman	1972-1974
Chairman	1974-1975
Awards Jury	1976

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF THE SECRETARY

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

ENTERGY NUCLEAR INDIAN POINT 2, LLC
ENTERGY NUCLEAR INDIAN POINT 3, LLC
ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3

Regarding the Renewal of Facility Operating Licenses
No. DPR-26 and No. DPR-64 for an Additional 20-year Period

NRC Docket Nos.
50-247 & 50-286

ASLB No.
07-858-03-LR-BD01

DECLARATION OF TIMOTHY B. RICE

Timothy B. Rice, hereby declares under penalty of perjury that the following is true and correct:

1. I am an Environmental Radiation Specialist II in the Bureau of Hazardous Waste and Radiation Management, Division of Solid and Hazardous Materials, of the New York State Department of Environmental Conservation (DEC). I currently serve as the New York State Project Lead for the State's participation in the ongoing investigation into radiological contamination of ground water under the Indian Point Generating Facility in Buchanan, New York, and the release of that contaminated water into the Hudson River. I submit this declaration to provide factual support to the contentions submitted by the State of New York.

Background

2. The Indian Point Nuclear Power Station consists of three nuclear reactors and spent fuel pools along with associated support systems, structures, and components. Between 1956

and 1969, the Atomic Energy Commission authorized Consolidated Edison Co. (ConEd) to construct the reactors, Units 1, 2, and 3. Unit 1 received a provisional operating permit in 1962, Unit 2 received its operating permit in 1973, and Unit 3 received its operating permit in 1975. ConEd operated Units 1 and 2, and the New York Power Authority operated Unit 3.

3. In 1974, the AEC ordered Unit 1 to cease generating power. Although Unit 1 no longer generates power, it "contains extensive common facilities that are required for the continued operation of Units 2 and 3." *See* October 1980 Decommissioning Plan for Indian Point Unit 1, p. 4 (included in the April 30, 2007 License Renewal Application).

4. In 2000 and 2001, Entergy Nuclear Operations, Inc. and its corporate affiliates (collectively Entergy) purchased Indian Point from NYPA and ConEd.

Spent Fuel Storage Pools

5. When the uranium can no longer be efficiently used to maintain the energy-generating fission process, it becomes spent fuel. To maintain efficient reactor performance, about one-third of the fuel is removed as spent fuel every twenty-four months, to be replaced with fresh fuel. The replacement occurs during planned outages.

6. After it is used in nuclear reactors to generate energy, spent nuclear fuel remains extremely hot and radioactive. The rods are therefore stored in specially designed water-filled spent fuel storage pools. The rods are placed in at least thirty feet of water which serves the dual purpose of acting as a radiation shield and dispersing the heat from the spent fuel. Spent fuel is moved from the reactor core via a transfer tube, which runs between the containment building and the fuel storage building, and into the transfer canal, which is part of the spent fuel pool. Attached as Figure 1 to this declaration is a diagram depicting the relative locations – for Unit 2 or Unit 3 – of the Hudson River, the cooling water intake structures, the turbine building, the

containment building, the transfer tube, and the spent fuel storage building, which houses the spent fuel pool.

7. Because no final disposal site has yet been developed, Indian Point's spent fuel has remained for decades in the facility's three temporary storage pools. In addition, the Nuclear Regulatory Commission has authorized IP2 and IP3 to increase the density of the spent fuel assemblies stored in their spent fuel pools.

Dry Cask Storage

8. When the spent fuel pools are full, the next step is to remove some of the spent fuel from the pools and move it to a permanent disposal site, such as the proposed Yucca Mountain site, or into dry cask storage. A specialized, heavy-duty crane assists in this transfer process.

9. Dry cask storage involves removing the spent fuel from the spent fuel pools and placing it in large above-ground cylinders. At present, spent fuel stored in the dry casks must be kept on-site at Indian Point because the federal government has yet to approve a permanent disposal facility at Yucca Mountain in Nevada, nor has an independent interim spent fuel storage facility been created, which is the only possible alternative.

Tritium and Strontium

10. Tritium and strontium are radionuclides. Tritium is radioactive hydrogen and is denoted as H-3. As with all ionizing radiation, tritium exposure increases the risk of developing cancer. Strontium-90, denoted as Sr-90, is chemically similar to calcium, and like calcium, it concentrates in bone. Exposure to strontium-90 has been linked to bone cancer, cancer in tissue near contaminated bone, and leukemia.

11. The EPA's current tritium drinking water concentration limit is 20,000 pCi/l; the drinking water standard for Sr-90 is 8 pCi/l.

12. At Indian Point, these radionuclides are one of the by-products of the fission process that takes place within the nuclear reactors. These radionuclides are present in the water that surrounds the fuel assemblies within the reactor, the transfer canal, and the spent fuel pools. Tritium is also present in various other plant systems at varying concentrations, including for example the reactor water storage tank and secondary steam lines.

Radioactive Materials Are Leaking from Indian Point's Spent Fuel Pools

13. Radioactive material is currently leaking from both the IP2 and IP1 spent fuel pools at Indian Point and is flowing in groundwater into the Hudson River. Plumes of strontium and tritium had been mapped under the facility.

14. In 2005, Entergy was excavating adjacent to the IP2 spent fuel pool to prepare for the installation of a new crane to support the dry-cask storage facility at Indian Point. In August 2005, Entergy discovered water leaking from a crack in the exterior of the IP2 concrete spent fuel pool.

15. In September 2005, Entergy determined that the leak from the IP2 spent fuel pool was contaminated with radioactive materials, including tritium. When the leak first occurred is not known. A review of records from the previous site operator, ConEd, has shown that a leak occurred in the stainless steel liner of the IP2 pool from 1990, when it was damaged by in-pool maintenance activities, until 1992, when the leak was discovered and was repaired. A subsequent Entergy inspection (2007) of the stainless steel pool liner in the Unit 2 transfer canal identified a small (1/8 to 1/4 inch) hole through the liner. Both the 1990 leak and the current hole in the liner are likely contributors to a groundwater plume currently flowing from IP2.

16. Concentrations of tritium from the IP2 spent fuel pool leak were detected early in the investigation of the ground water in the monitoring wells closest to the IP2 spent fuel pool at

levels as high as 600,000 pCi/l, thirty times the drinking water standard of 20,000 pCi/l. At this time, the maximum concentrations of tritium detected in the monitoring wells closest to the Hudson River have not exceeded twenty-five percent of the drinking water standard. However, this does not guarantee that the concentrations reaching the river will not increase over the 20-year extension of IP2 if additional leaks occur in the spent fuel pools or other contaminated systems.

17. As part of the investigation prompted by the IP2 spent fuel pool leak, Entergy also determined that radioactive strontium-90 is leaking from the IP1 spent fuel pool. The investigation also has revealed that other radioactive constituents, including a range of fission and activation products including cesium, cobalt, and nickel, are being released from the IP1 spent fuel pool. Strontium-90 has been detected migrating in groundwater. The other constituents remain in the soil and rock in the vicinity of IP1.

18. The IP1 leak has existed since some time before 1994 when the leak was initially discovered. Though a lack of data makes it difficult to determine exactly how long it has existed, it is likely that it has been ongoing throughout much of the life of the plant. ConEd, which owned the site in 1994, took corrective actions at that time to minimize and contain the leaks and it believed those actions were effective. However, the groundwater sampling performed over the past two years has demonstrated that not all of the ongoing IP1 spent fuel pool leakage is being contained. In fact, some of the contaminated groundwater has been, and continues to be, bypassing the IP1 drain systems that ConEd had stated were containing the contamination.

19. Concentrations of strontium from the IP1 spent fuel pool leak have been detected at almost fourteen times the drinking water standard of 8 pCi/l at the monitoring well closest to the IP1 spent fuel pool. Concentrations of Sr-90 at a monitoring well close to the Hudson River have

been detected at approximately 3.4 times the drinking water standard.

20. As noted above, in addition to tritium and strontium, other radiological contaminants, including cesium, cobalt, and nickel, have been detected in groundwater as a result of leaks from the IP1 spent pool.

21. As has been demonstrated at other facilities having radionuclide-contaminated soil and bedrock, the presence of these radioactive contaminants beneath and around Indian Point site structures will likely increase the cost and extent of the eventual decommissioning of the reactor facilities, particularly IP1. A nearby example of this impact was the decommissioning of the Cintichem medical-radioisotope production reactor and hotlab in Tuxedo, NY. There, the presence of radioactive contaminants in the underlying bedrock resulted in major extensions of the decommissioning schedule and a significant increase in costs due the significant efforts needed to remove this contamination.

Additional Systems Contain or Convey Radioactive Nuclides

22. In addition to the spent fuel pools, transfer canals, and reactor, other systems, structures, and components at Indian Point contain or convey radionuclides. As the NRC recognized in 2006: "Any system containing liquids which originated or have a connection with reactor coolant have the potential to contain tritium. Examples are the spent fuel pool, liquid radwaste storage tanks, refueling water storage tanks, condensate storage tank, turbine sumps, and steam generator blowdown lines." *See* March 2006 NRC Talking Point slide entitled "Tritium at Nuclear Power Plants in the United States; Slide 3: Background" (attached to this declaration as Figure 2).

23. On April 24, 2007, during a routine twice weekly conference call update to the NRC and DEC in which I participated, Entergy disclosed that on or about April 7, a secondary steam

pipe buried approximately four feet underground and running between IP2 and IP3 began leaking. This line contains steam that, when condensed to water, would measure between 2,000 and 3,000 pCi/l of tritium. It was detected by the presence of steam venting upwards through the soil and pavement near the unit 3 turbine building. Entergy also presented this information to local, state, and federal stakeholders during a special conference call (to report an unrelated incident) held on April 25.

Future Environmental Impacts

24. The groundwater investigation that has been under way since late 2005 has shown that any contamination entering the groundwater in the vicinity of the IP1 and IP2 spent fuel pools and reactor containment buildings will, dependant upon a particular contaminant's ability to bind to soil and rock surfaces, eventually end up in the Hudson River. This occurs because groundwater flow in the areas occupied by IP1, IP2, and IP3 is from the east, north, and south, and to the west towards the river. This water flow towards the river takes place in all underlying materials saturated with groundwater, including soils, construction fill, and fracture flow in the underlying bedrock.

25. Entergy inspected accessible portions of the stainless steel liner in the Unit 2 spent fuel pool and the liner of the Unit 2 transfer canal. The full extent of the leaks in the IP2 spent fuel pool is not known. Unless all of the spent fuel stored in the IP2 spent fuel pool can be removed, it is unlikely that the balance of the liner can be adequately inspected. As long as the IP2 reactor continues to operate, it will not be possible to remove the fuel to allow a complete inspection of the IP2 spent fuel pool liner. The inability to inspect a large portion of the liner will prevent Entergy from definitively concluding that no other leaks in the IP2 spent fuel pool exist.

26. While removal of the spent fuel from the IP1 spent fuel pool will remove the active source of contaminants from that structure, significant amounts of previously-released and accumulated radionuclides will remain within the fractures in the concrete structures, and in the underlying soil, fill, and bedrock, and will continue to act as a source of groundwater contamination for many years into the future.

Conclusion

27. In sum, Indian Point is leaking radioactive material from two of its spent fuel pools into groundwater on-site, which is in turn discharging into the Hudson River. When most of these leaks began is unknown. Although Entergy is planning to remove all of the spent fuel from the IP1 spent fuel pool and place it into dry cask storage, that is not true for the IP2 spent fuel pool, which will need to continue to accept new spent fuel in order for the Unit 2 reactor to continue operating. Even if Entergy's attempts to stop future leaks in IP1 and IP2 are completely successful, this cumulative historical contamination in the structures themselves and in underlying soil, fill, and bedrock will act as a reservoir that will continue to release contaminants into the groundwater for the foreseeable future.

Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated: Albany, New York
November 26, 2007

Timothy B. Rice

TIMOTHY B. RICE

Rice Declaration Attachment 1

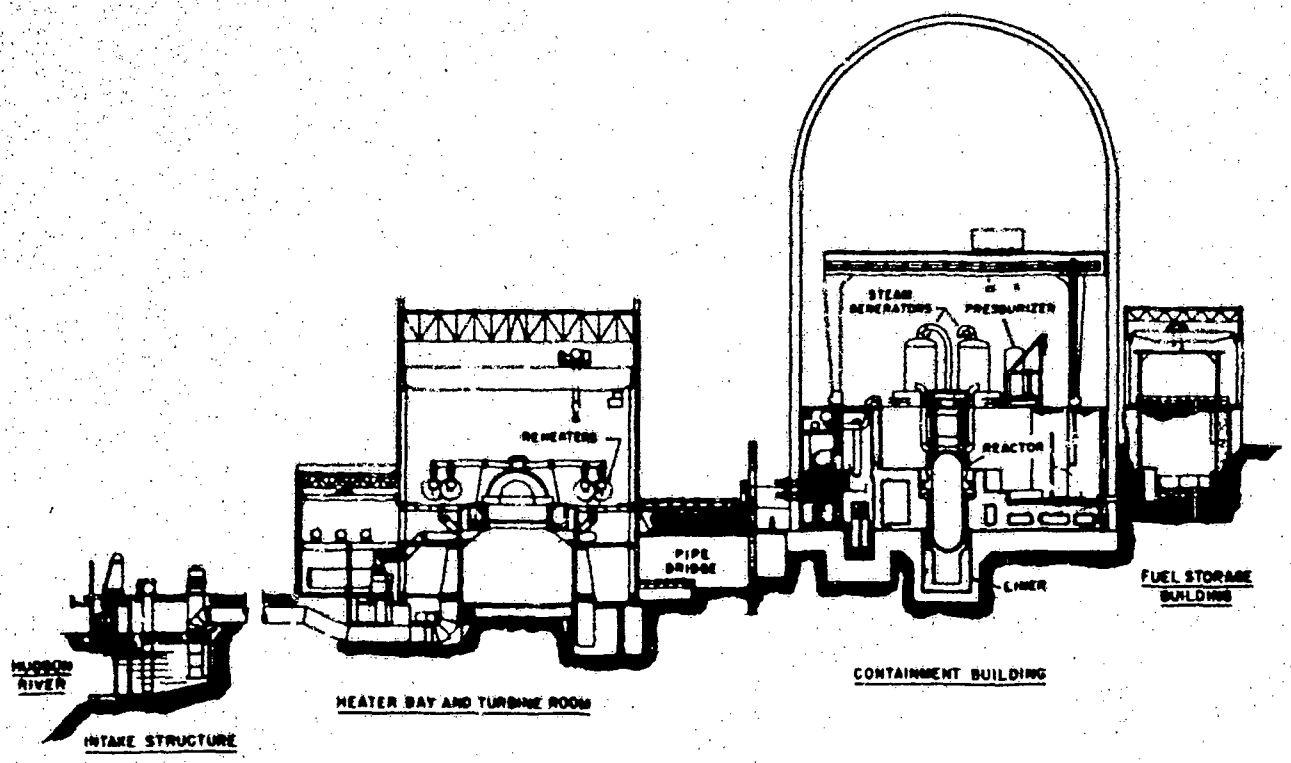

Figure 1.2-4 Cross Section of Plant

Tritium at Nuclear Power Plants in the United States

Slide 3: Background

Indian Point
Unit 1 has been shutdown since 1974, but the spent fuel pool still contains fuel and radioactive water. Tritium, nickel-63, cesium-137, strontium-90, and cobalt-60 have been detected onsite at Indian Point. It is suspected that these are the result of leakage from the unlined Unit 1 spent fuel pool. Indian Point is operating a filter/demineralizer in the Unit 1 spent fuel pool to reduce the concentration of radioactive material that may continue to leak.

Unit 2 spent fuel pool leakage was discovered in August 2005 during excavation work near the fuel storage building loading bay. Two cracks were discovered along the wall of the spent fuel pool with leakage. Analysis indicated that the material had the same radiological and chemical properties as spent fuel pool water. In September 2005, an onsite monitoring well was sampled and returned positive results for tritium.

Braidwood
In March 2005, the Illinois Environmental Protection Agency notified the licensee of reports of tritium in wells in the nearby community. In November 2005, the NRC was notified that elevated levels of tritium had been measured in groundwater monitoring wells at Braidwood at levels up to 58,000 pCi/L. This was attributed to contamination from historical leakage of vacuum breaker valves along the circulating water blowdown line. The line is routinely used for radioactive liquid releases to the Kankakee River. At Braidwood, the line is about 5 miles long and contains 11 vacuum breaker valves, spaced along the length of the line.

Braidwood investigation found that significant unplanned radioactive releases from three of the 11 vacuum breaker valves occurred during 1996, 1998, and 2000. Each releases from the vacuum breaker valve occurred during a period coincident with ongoing, liquid radioactive releases through the blowdown line, resulting in tritium entering the groundwater system in the vicinity of the leaking vacuum breaker valve. Groundwater samples were taken onsite and offsite and tritium levels were detected as high as 225,000 to 250,000 pCi/L.
1996 event – 250,000 gallons
1998 and 2000 events – 3,000,000 gallons each

Between March 2005 and March 2006, Exelon sampled the water in drinking water wells of nearby homeowners. Tritium levels between 1,400 and 1,600 pCi/L were identified in one residential drinking water well.

Sources of Tritium at Nuclear Power Plants
Any system containing liquids which originated or have a connection with reactor coolant have the potential to contain tritium. Examples are the spent fuel pool, liquid radwaste storage tanks, refueling water storage tanks, condensate storage tank, turbine sumps, and steam generator blowdown lines.

Congressional Reaction
New York Senator Hillary Clinton and Illinois Senator Barack Obama have been highly interested in the status of groundwater contamination at Indian Point and Braidwood, as well as other nuclear power plants in the state they represent.

Senator Obama introduced legislation requiring nuclear power plants to quickly inform state and local officials of accidental or unintentional leaks of radioactive substances. The bill passed the Senate Environmental and Public Works Committee on September 13, 2006.

Public outrage and news coverage

Public meetings have been held at both Indian Point and Braidwood.

Due to intense public interest from local officials and resident, Exelon held three public information forums: a public information meeting was sponsored by the village of Godley, IL; a public meeting was sponsored by U.S. Senator Richard Durbin; and a meeting with local officials was organized by U.S. Representative Jerry Weller.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Operations, Inc.

Docket Nos. 50-247-LR, 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

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DECLARATION OF PETER A. BRADFORD

Peter A. Bradford, hereby declares under penalty of perjury that the following is true and correct:

1. My name is Peter Amory Bradford. I live in Peru, Vermont. My resume is attached to this declaration.

2. I am President of Bradford Brook Associates, a firm advising on utility regulation and energy policy. I teach a course entitled "Nuclear Power and Public Policy" at Vermont Law School. I have been a member of the Keystone Center "Nuclear Power Joint Fact Finding" (June, 2007) and the National Research Council of the National Academy of Sciences' Committee on "Alternatives to the Indian Point Energy Center for Meeting New York Electric Power Needs" (June, 2006). I was also a member of the International Expert Panel advising the European Bank for Reconstruction and Development assessing the economic case for completing Khmelnitsky 2 and Rovno 4 (K2/R4) – two partly built, Russian designed 1,000 MW VVER nuclear units in Ukraine – to replace the two operational 1,000 MW units at Chernobyl (February, 1997).

3. I have chaired the New York Public Service Commission (1987-95). In that capacity, I was an *ex officio* member of the New York State Energy Planning Board.

4. I served on the Maine Public Utilities Commission (1971-1977 and 1982-87) and was Chairman in 1974-1975 as well as 1982-87.

5. I served as a member of the U.S. Nuclear Regulatory Commission (1977-82).

6. The Nuclear Regulatory Commission's approach to assessing alternatives to the construction and operation of nuclear power plants has been deficient since the agency was created in 1975. In particular, the NRC has been ineffective in assessing the role that energy efficiency can play (and has played) in displacing nuclear power plants.

7. Nearly half of all of the more than 200 plants licensed for construction by the NRC in its history have been cancelled, often after expenditures of millions and sometimes billions of dollars. Many others were delayed long past their scheduled completion dates, dates by which the NRC (or its predecessor, the Atomic Energy Commission) found that they would be needed to meet demand for electricity. Another dozen plants have been prematurely closed, some on short notice. In most cases, the licensee cited absence of need as a primary reason for the cancellation or deferral. In very few cases was a central generating facility of equivalent capacity constructed to replace the cancelled capacity. No significant power shortage has resulted from these cancellations, deferrals, or closings.

8. A study done for me when I was an NRC Commissioner in 1979 concluded, *inter alia:*

> The Commission has consistently failed to perform full cost-benefit analyses for reasonable alternatives as required by NEPA. Alternatives other than coal are routinely dismissed with boilerplate language in environmental impact statements. Commission estimates always favor nuclear over coal and a NFP (need for power) determination is always made affirmatively. NRC environmental statements display a clear bias in favor central station facilities, and a mix of potentially more cost-effective (and environmentally benign) technologies is never adequately assessed.

Gerald Warburg, "A Study of NRC Procedures for Assessing Need for Power and Alternative Energy Sources in Fulfillment of the NEPA Requirements for Environmental Impact Statements" (1979).

9. The Environmental Report in this proceeding reflects the flaws in the NRC's historic approach to assessing alternatives to the operation of a nuclear plant. In so doing, the applicant seems to be relying on the NRC to accept its flawed analysis despite the NRC's own GEIS requirements to analyze combinations of efficiency and renewables. Not only does the applicant confine the alternatives

analysis to central generating facilities but – by assuming the operation of the two Indian Point units – it assumes away the urgency that has demonstrably been the most effective spur to large scale energy efficiency programs. See Entergy Environmental Report, § 7.0 to 7.5, pp7-1 to 7-5.

10. The National Academy of Sciences panel on alternatives to the continued operation of one or both Indian Point units – while taking no position as to whether Indian Point should continue to operate – concluded:

> A wide and varied range of replacement options exists, and *if a decision were definitely made to close all or some part of Indian Point by a date certain,* the committee anticipates that a technically feasible replacement strategy for Indian Point would be achievable .... [F]rom the committee's analysis, no "right" or clearly preferable supply alternative to Indian Point emerged. A replacement strategy for Indian Point would most likely consist of a portfolio of the approaches discussed in this report, including investments in energy efficiency, transmission, and new generation.

"Alternatives to the Indian Point Energy Center for Meeting New York Electric Power Needs," the National Research Council, June, 2006, p. 3 (emphasis added).

11. The recent history of the electric power industry in the United States demonstrates beyond dispute the ability of a large power system such as New York effectively to create portfolios of replacement energy resources once a decision has clearly been made to close a particular unit or once unexpected circumstances produce the same result. Consider the following examples:

A. The 820MW Shoreham nuclear power plant on Long Island was – until 1988 – included in the Long Island Lighting Company's plans for meeting its load from mid-1989 onward. Late in 1988, LILCO and the State of New York agreed that the plant should not operate, and the settlement was affirmed by state regulators and the utility's board of directors by June 1989.

   Like the downstate New York region today, Long Island's ability to import power faced substantial transmission constraints. Shoreham's percentage of the LILCO system peak was greater than that of the two Indian Point units in the Lower Hudson River Valley, New York City and Long Island. Many in the electric industry, in the federal government, and in the media forecast serious power shortages on

Long Island in the years following the agreement not to operate the plant.

Once the question of Shoreham's future was clear, LILCO and the State moved rapidly to put together a replacement power program consisting of demand side management, load management, targeted maintenance to assure high availability of other plants at peak times, transmission upgrades, peaking units, and independent power production, some of it renewable.

Though LILCO operated below its reserve requirement for two or three summers after the Shoreham settlement, power supply was at all times adequate.

Through load management programs alone, LILCO gained control of 130MW of its potential load before the 1989 summer peak.

B. In 1986, the State of Maine and its utilities reached an agreement to end Maine's involvement in the Seabrook nuclear power plant. At the time of this agreement, Seabrook was expected online within two years, which would have meant about 110 megawatts for the three Maine companies. In the years preceding the agreement, Maine had pioneered in the use of competitive bidding for new power resources and had come to realize that the amount of renewable resources – specifically biomass – to be had was far greater than had been forecast in the early 1980s.

The Seabrook power was replaced almost entirely by biomass energy from Maine's forests, with substantial economic advantages to Maine electric customers, taxpayers, wood owners, and workers. These biomass plants would not have been built had Maine remained in Seabrook. They were built to meet the market opportunity created by Maine's decision to get out of Seabrook. A subsequent study showed substantial economic benefit to Maine from the decision to disengage from Seabrook.[1]

C. In June 1989, the voters of Sacramento, California voted to close the Rancho Seco nuclear power plant, which supplied 913 of the Sacramento Municipal Utility District' s (SMUD) 2,100 MW load. Using purchased power to bridge the gap, SMUD embarked on a

[1] "Energy Choices Revisited: An Examination of the Costs and Benefits of Maine's Energy Policy", a study for Mainewatch Institute by Economic Research Associates, the American Council for an Energy Efficient Economy and the Tellus Institute, 1994.

program of extensive energy efficiency coupled with cogeneration, renewable energy and purchased power. In hindsight, this program – which clearly would not have happened had the nuclear plant remained in operation – has worked out to the advantage of the Sacramento community.[2]

D. Between mid-2000 and mid-2001, the state of California was repeatedly threatened with power shortages and did indeed experience blackouts. However, by the summer of 2001, load management and demand side management programs of various sorts had produced several thousand megawatts in savings above and beyond what had been expected from the California efficiency programs that had been in place a year earlier.[3] These rapidly assembled efficiency resources, many of which remain in place, were largely responsible for bringing the California energy crisis to an end and for keeping the lights on until power purchases, new power plant construction and an end to market manipulation restored the state to a more lasting equilibrium.

12. In each of the foregoing cases, the amount of energy efficiency and other resources put into place vastly exceeded the forecasted availability of a few years earlier. It is the realization that generating capacity will not be available that creates the climate in which alternative resources will be developed and put into place. For sound economic and political reasons, the planning and investment necessary to add large blocks of replacement energy efficiency, purchased power, transmission or new generation to a system will not occur without a clear indication that the investments are needed and have a reasonable likelihood of earning a competitive return.

13. Any claim that a decision to extend the license of the two Indian Point units is merely a decision to keep the Indian Point option and need therefore not be regarded as an either/or decision between the nuclear plants and a decision to

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[2] SMUD's history states, "To replace nuclear power, the SMUD Board moved away from the concept of a large central plant toward diverse power sources, such as cogeneration plants, wind power, low-cost purchased power from the Pacific Northwest and Canada, and research and development of renewable resources and advanced technologies like solar, fuel cells, gas turbines and biomass." SMUD's history: 1990s: Moving Into Leadership on Green Energy, Conservation, *available at* http://www.smud.org/about/history-1990s.html (last visited Nov. 27, 2007).

[3] "In the summer of 2001, California's energy efficiency programs and energy conservation–related efforts saved between 3,200 and 5,600 MW and reduced peak demand by an average of 8 percent, which helped the state avert 50 to 160 hours of rolling black outs." Goldman, C., J. Eto, and G. Barbose, "California customer load reductions during the electricity crisis: did they help to keep the lights on?" LBNL-49733. (2002) (available at http://eetd.lbl.gov/ea/EMS/reports/49733.pdf), cited in "Energy Efficiency: California's Highest Priority Resource", California Public Utilities Commission and California Energy Commission, June 2006, at 4).

replace them with other resources ignores the realities of power supply planning and procurement. In order to comply with its NEPA obligations the NRC needs an analysis that reveals whether other options are environmentally preferable to extending the Indian Point license. The agency and the licensee cannot discharge this responsibility just by putting the Indian Point units forth as options and trusting to other jurisdictions that the optimal course will be chosen. As the above examples show, it is the realization that the expected generation source will not be available or ought not to be used that brings about the conditions under which the demand side management and renewable alternatives are able to replace them. Only an analysis fully consistent with power supply procurement realities – including the abundance of available energy efficiency and the conditions necessary to bring it into being – will enable the NRC to assess the environmental impacts of its decision on relicensing the Indian Point units.

14. Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated: November 28, 2007
Peru, Vermont

Peter A. Bradford

RESUME OF PETER A. BRADFORD

Peter Bradford advises and teaches on utility regulation, restructuring, nuclear power and energy policy in the U.S. and abroad. He has been a visiting lecturer in energy policy and environmental protection at Yale University and has taught courses entitled "Nuclear Power and Public Policy" and "The Law of Electric Utility Restructuring" at the Vermont Law School. He has recently served on a Keystone Center fact finding collaboration on nuclear power and a National Academy of Sciences panel evaluating the alternatives to continued operation of the Indian Point Nuclear Power Plants in New York. He is also affiliated with the Regulatory Assistance Project, which provides assistance to state and federal energy regulatory commissions regarding economic regulatory policy and environmental protection. He is vice-chair of the Board of the Union of Concerned Scientists.

He served on a panel advising the European Bank for Reconstruction and Development on how best to replace the remaining Chernobyl nuclear plants in Ukraine and also on an expert panel advising the Austrian Institute for Risk Reduction on regulatory issues associated with the opening of the Mochovce nuclear power plant in Slovakia. He advised the Vermont Legislature on issues relating to spent fuel storage at Vermont Yankee and the Town of Wiscasset, Maine, on issues related to the storage of spent nuclear fuel at the site of the former Maine Yankee nuclear power plant.

He has advised on restructuring issues in many states and has testified on aspects of electricity and telecommunications restructuring in many U.S. states.

He has advised on energy, telecommunications and water utility restructuring issues in China, Armenia, Azerbaijan, Georgia, India, Indonesia, Mongolia, Canada, Russia, South Africa, and Trinidad and Tobago. He is a member of the Policy Advisory Committee of the China Sustainable Energy Program, a joint project of the David and Lucille Packard Foundation and the Energy Foundation.

He chaired the New York State Public Service Commission from 1987 until 1995 and the Maine Public Utilities Commission from 1982 until 1987. During these years, New York resolved its stalemate over the Shoreham nuclear power plant and Maine resolved its similarly controversial involvement in Seabrook, both on favorable economic terms. He was Maine's Public Advocate in 1982 and was President of the National Association of Regulatory Utility Commissioners during 1987.

He served on the U.S. Nuclear Regulatory Commission from 1977 until 1982. During his term, the NRC undertook major upgradings of its regulatory and enforcement processes in the wake of the Three Mile Island accident.

Prior to becoming a member of the NRC, he had served on the Maine Public Utilities Commission (1971-1977) and was Chairman in 1974-1975.

Mr. Bradford was an advisor to Maine Governor Kenneth Curtis from 1968 to 1971, with responsibilities for oil, power, and environmental matters. He assisted in preparing landmark Maine laws relating to oil pollution and industrial site selection and was Staff Director of the Governor's Task Force on Energy, Heavy Industry and the Coast of Maine.

Mr. Bradford is the author of Fragile Structures: A Story of Oil Refineries, National Security and the Coast of Maine, a book published by Harper's Magazine Press in 1975. His articles on utility regulation and nuclear power have appeared in many publications, including The New York Times, The Washington Post, The Los Angeles Times, The Boston Globe, Newsday, and The Electricity Journal.

He is a 1964 graduate of Yale University and received his law degree from the Yale Law School in 1968.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Operations, Inc.

Docket Nos. 50-247-LR, 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

DECLARATION OF LYNN R. SYKES

Lynn R. Sykes, Ph.D. hereby declares under penalty of perjury that the following is true and correct:

1. I am currently the Higgins Professor Emeritus, Earth & Environmental Sciences at the Lamont-Doherty Earth Observatory of Columbia University.

2. During the course of my career, I have studied seismic issues in the throughout the United States and the world. Among the areas I have studied is New York City Seismic Zone (which includes portions of New York State, New Jersey, Pennsylvania, and Connecticut). My CV is attached to this declaration.

3. I have prepared a report concerning earthquake activity in intraplate continental regions such as eastern North America, with emphasis on issues directly relevant to earthquake hazard in the greater tri-state New York City

Seismic Zone and the area in and around the site for the Indian Point Nuclear Power Station. Among other things, the report concludes that: (1) the application for license renewals for IP2 and IP3 contains very dated information on earthquake hazards; (2) the application for license renewals for IP2 and IP3 underestimates earthquake hazard ; (3) updated information on instrumentally-recorded earthquakes is vital to assessments of earthquake hazards to Indian Point ; and (4) earthquake risk as well as hazard need to be considered in safety analyses for Indian Point.

4. The report and CV are true and correct to the best of my personal knowledge.

5. Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated:

November 29, 2007
Palisades, New York

Lynn R. Sykes

Statement in Support of New York State Contentions and in Response to the April 30, 2007 License Renewal Application Submitted by Entergy for Indian Point Units 2 and 3

by Lynn R. Sykes, Ph.D.
Higgins Professor Emeritus, Earth & Environmental Sciences
Lamont-Doherty Earth Observatory
of Columbia University, Palisades NY 10964

November 29, 2007

Lynn R. Sykes, Higgins Professor Emeritus, Earth & Environmental Sciences at the Lamont-Doherty Earth Observatory of Columbia submits the following statement in support of the contentions submitted by the State of New York in response to the April 30, 2007 license renewal application filed by Entergy Nuclear Operations, Inc., and its corporate affiliates for the Indian Point Power Station located in the Village of Buchanan in Westchester County.

I. The Application for License Renewals for IP2 and IP3 Contains Very Dated Information on Earthquake Hazards

IP3 FSAR Update (2007) Ch. 2.8 lists a table from an unpublished final report on the Indian Point seismic network for the period 1975-1990, but information on earthquakes since 1982, especially that for the Ardsley NY shock of 1985, is not included in either it or IP2 FSAR Update (2007). Table 2.8-1 in that IP3 FSAR Update does not list any earthquakes more recent than 1979. That document also repeats a quote in section 2.8.1 from J. Lynch, who made a study for Indian Point 1 more than 45 years ago, ". . that the probability of a serious shock occurring in this area for the next several hundred years is practically nill [sic]. The area therefore would certainly seem to be as safe as any area at present known." It also quotes him as saying, "estimated maximum ground acceleration of 0.03 g is reasonably conservative for the area." It is not clear from the FSAR Update (2007) for IP1 what design acceleration was used for IP1.

None of the references on earthquakes in Ch. 2 pp. 115-116 of that IP2 FSAR Update are more recent than 1982; only one in IP3 FSAR Update (2007) in Ch. 2.8 (the above unpublished final report) is more recent than 1986. All of these are marked in green as historical information. That Update lists the use of four sets of strong motion data from 1934 to 1952 and on pp. 14 and 15 of Ch. 16 states "no strong motion records were available for the Eastern United States. . ." That statement is very outdated since strong-motion data have existed for the eastern and central United States and Canada for many years as well as for other similar intra-plate regions. Such statement would also appear to be inconsistent with the NRC's recently-initiated analysis of Generic Issue 199 relating to seismic hazards in the central and eastern United States.

II. The Application for License Renewals for IP2 and IP3 Underestimates Earthquake Hazard

A. Can Future Damaging Earthquakes at Indian Point be Excluded?

IP2 FSAR Update (2007) states in both Ch. 1, p. 3 and Ch. 2, p. 1, "Seismic activity in the Indian Point area is limited to low-level microseismicity." The IP2 FSAR Update (2007, Ch. 2, pp. 104-119) includes reference to the Woodward-Clyde Consultants report of 1982 that, in turn, states on p. 113, "Earthquakes occurring near Indian Point have been characterized as shallow focus (<10 km) and low magnitude (1.0-3.0) . . ." While these statements are correct if they refer to the earthquakes during the short period of observations only, they are misleading about the potential for damaging earthquakes at Indian Point according to a broad consensus on how to interpret available data.

Figure 1 is the eastern half of one of several earthquake hazard maps for the United States published and updated periodically by the U. S. Geological Survey (USGS). These maps are derived from the observed earthquake distribution. Figure 1 shows contours of calculated horizontal ground acceleration that would be exceeded with 2% probability in 50 years at a frequency of 5 cycles per second (5 Hz.) for 5% of critical damping. Probabilistic maps of this type are used widely and for building codes and setting insurance rates. Some may argue that critical structures such as bridges, hospitals, and nuclear power reactors should be designed for even higher accelerations and/or less likely exceedance in 50 years. The earthquake hazard map prepared by the USGS (Figure 1) shows that southeastern New York State and northern New Jersey are characterized by a concentration of higher values compared to those of many other areas of the eastern and central United States. Lynch's very old depiction of the Indian Point site as being "as safe as any area at present known" and the repeat of that statement in the IP3 FSAR Update of 2007 clearly are not consistent with the "bulls eye" of higher accelerations in Figure 1, the current USGS earthquake hazard map. Lower values can be seen for central New York State, Michigan, Florida and much of Pennsylvania. Higher values are shown near Charleston, South Carolina, New Madrid, Missouri, and northern New York State and comparable values in southern New Hampshire. Nevertheless, those four areas have much smaller populations and assets at risk than the greater New York City region.

In his discussion of seismic regionalization in the contiguous United States, Richter

(1959) places southeastern New York in a broad region of intensity VIII. He goes on to state "*New York City* should be studied in great detail from the point of view of microregionalization. It is within the range of probable VIII on average ground from a great St. Lawrence earthquake and the shock of 1884 confirms the presence of a local source, probably offshore, also capable of producing VIII." For southeastern New York the USGS probabilistic seismic hazard maps (e.g. Frankel and others, 2005), unlike NRC seismic requirements from the 1970s, contain a contribution to hazard at Indian Point from large earthquakes in the St. Lawrence Valley.

While higher accelerations are calculated by USGS for nearly all of California, nuclear power reactors and other critical structures there are designed for higher accelerations than those at Indian Point. Seismic hazard estimates for some of the first nuclear power reactors in the eastern and central United States, such as that by Lynch, seem to have had in mind a comparison with some of the world's most active earthquake areas, such as those along plate boundaries in Japan, Alaska, Chile, and Peru. By contrast, the eastern two-thirds of United States is an intraplate region of lower, but not negligible, earthquake hazard. The design safe-shutdown acceleration for Indian Points 2 and 3 is 0.15 g. The Diablo Canyon reactors in California were designed for higher accelerations. Seven adjacent nuclear power plants in Niigata prefecture, Japan, were subjected to high accelerations during the earthquake of July 16, 2007 of magnitude 6.6, which occurred nearly beneath them (EERI, 2007). The Niigata reactors were designed for accelerations of only about 0.17 to 0.27g. Four of the observed horizontal accelerations exceeded design values by factors of 2.0 to 3.6. The same plate boundary was the site of the damaging and large Niigata earthquake of 1964 of magnitude, M, 7.5. Even higher accelerations could have occurred if the 2007 earthquake had been as large as that of 1964.

B. Significant Historic Earthquakes and their Relationship to Geologic Terranes

The term microseismicity, as quoted above from the updated FSAR, often is used by seismologists to refer to earthquakes of magnitude smaller than 3.0 (Mogi, 1985, pp. 67-69), which are rarely felt in California and Japan. Many earthquake as small as magnitude 2, however, are routinely felt in the lower Hudson Valley and in northern New Jersey. Shaking of intensities either V or VI has been reported for many earthquakes smaller than M 3.0 (Sykes and others, 2007). Richter (1958, p. 16) states, "*Microseismic* effects are small-scale, observable

only with instruments."

Figure 2 indicates that 28 earthquakes of M 3 or greater are known to have occurred in the greater New York City-Philadelphia area. The record for events of M 3.0 is complete since about 1928, that for M 3.5 since about 1840, and that for M 5 and larger since 1737 (Sykes and others, 2007). Several earthquakes larger than M 4.7 in the area of Figure 2 and in nearby Pennsylvania have caused damage. *These events are not microearthquakes*. Their occurrence is, in fact, the basis for the region of higher accelerations in Figure 1 and on other USGS maps of earthquake hazard.

A number of the events in Figure 2 occurred within older hard rocks of the Manhattan Prong, the geologic province in which Indian Point is located. Included are the earthquakes of 1848 of M 4.35, 1985 of 4.1, 1845 of 3.75, 1874 of 3.5, and perhaps the poorly located shock of 1737 of M 5.1. The Reading Prong-western Hudson Highland geologic province is located within a few kilometers of Indian Point and includes earthquakes in 1951 of M 3.85, 2003 of 3.5, 1957 of 3.25 and perhaps two poorly located shocks of 1783 of M 5.1 and 4.65. Most of the other earthquakes in Figure 2 occurred beneath the thin coastal plain sediments of New Jersey and just offshore of New York City in what are inferred to be older hard rocks (Sykes and others, 2007). The Manhattan Prong and the Reading Prong-western Hudson Highland geologic provinces and their associated past earthquake activity are in contact near Indian Point. Farther southwest they are separated by the region of lower activity in Figure 2 beneath the younger and mostly weaker rocks of the Newark basin.

III. Updated Information on Instrumentally-Recorded Earthquakes is Vital to Assessments of Earthquake Hazards to Indian Point

Lamont-Doherty Earth Observatory of Columbia University in conjunction with several local institutions has operated a network of three or more seismograph stations in the greater New York City area since 1962. Coverage by a more extensive network, which has evolved with time, extends from 1974 to the present. Instrumentally-recorded earthquakes from 1974 through 2006 are shown in Figure 3 (for the same area as in Figures 2). Events since 1974 are located more precisely than nearly all of those based solely on intensities (i.e. felt reports of shaking and damage). Furthermore, since 1974 instruments have detected many smaller events, i.e., about 71% of the total known earthquakes from 1677 to 2004 in Figure 4. Consequently, more and

better data are available now than approximately 30 years ago when earthquake hazards in the greater New York City region and the earthquake safety of the Indian Point nuclear power plants were first debated.

Instrumental locations are relevant to a finer definition of the distribution of earthquakes than either those shown in Figure 2 or data used to calculate accelerations in Figure 1. Events in Figure 3 are purposely shown free of other geologic information so as to portray the detailed spatial distribution of earthquakes more clearly. For several decades the location capability of the local seismic network was strongest for events in northern New Jersey and southeastern New York State. Coverage south and southwest of New York City in Figure 3 has not been as good.

We find that earthquakes in Figure 3 originate from many faults rather than a few single major faults. Nevertheless, earthquake activity is not distributed uniformly throughout that area, but is concentrated in prominent zones, such as the Ramapo seismic zone (RSZ) in the eastern part of the Reading Prong where station coverage has been strongest since 1974. The southeastern boundary of that 12-km wide zone, which is nearly vertical, extends from near the surface trace of the Mesozoic Ramapo fault to depths of 12 to 15 km. Earthquakes in that zone are occurring within older rocks. Ratcliffe (1980) states that current seismic activity along the Ramapo zone may be more strongly controlled by the presence of through-going crustal structures than it is by more superficial Mesozoic faults. Which faults within the Ramapo seismic zone are active is not clear and remains controversial. Earthquake activity in the Manhattan Prong also extends to depths of 12 to 15 km (Sykes and others, 2007).

A new result based on 34 years of instrumental data is that activity in the Manhattan Prong cuts off abruptly along a nearly vertical, northwest-striking boundary between B and B' in Figure 3 that extends from Stamford Connecticut to Peekskill New York (locations in Figure 4). Activity in Figure 3 is absent to the northeast of that line in the eastern Hudson Highlands. This boundary is sub-parallel to the youngest brittle faults in the Manhattan Prong. One of them, the Dobbs Ferry fault was the site of the 1985 shock of M 4.1. The Peekskill- Stamford seismic boundary is inferred to be a similar and perhaps a more through-going fault or fault zone.

An abrupt bend in the Hudson River to a northwesterly trend is situated near line B-B' (Figure 3), which is close to Indian Point. Fisher et al. (1976) and Ratcliffe (1976) indicate a northwesterly-striking fault on the north side of that segment of the River. An extension of that trend to the northwest follows a major lineament that crosses the Hudson Highlands on the

Preliminary Brittle Structures Map of New York (Isachsen and McKendree, 1977). Most of the seismic activity along the Ramapo seismic zone and the Peekskill-Stamford line appears to end at or near their intersection.

Two well-located events at depths of 15 km are situated at the intersection of the Ramapo seismic zone and the Peekskill-Stamford line just to the northwest of Peekskill near Annsville. Seborowski et al. (1982) conclude that epicenters of a shallow earthquake sequences near Annsville from 1977 to 1980 are aligned northwesterly. That trend indicates that they were situated along a fault or faults at or near the Peekskill-Stamford boundary. Seborowski et al. (1982) and Quittmeyer et al. (1985) of the Woodward-Clyde consulting firm obtained focal mechanism solutions for two small events and composite solutions for two earthquake sequences that occurred within the Indian Point seismic network near Peekskill. The solutions involve a predominance of thrust faulting along nodal planes striking NW to NNW. The strikes of nodal planes of three of those mechanisms are compatible with slip along the Peekskill-Stamford line. These Woodward Clyde studies, however, which appear to have been narrowly focused on the Mesozoic Ramapo fault, were based on a limited data set, and were mostly restricted to the immediate vicinity of Indian Point. They neither reported the Peekskill-Stamford seismic boundary nor considered hazards related to the totality of earthquake activity either near Indian Point or within the Manhattan Prong and the Reading Prong-western Hudson Highlands.

Indian Point is situated at the intersection of the two most striking linear features marking earthquake activity in Figure 3 and also in the midst of a large population that is at risk in case of an accident to the nuclear plants. This is clearly one of the least favorable sites in Figure 3 from an earthquake hazard perspective.

Present knowledge about the state of stress in southeastern New York and northern New Jersey (Sykes and others, 2007) indicates that maximum compression in the crust of the earth is nearly horizontal and is oriented about N64^{o}E (Fig. 2). That orientation can facilitate the occurrence of earthquakes of mainly strike-slip type along brittle faults trending northwesterly, as in the 1985 Ardsley earthquake (Seeber and Dawers, 1989) and along brittle faults oriented about NW to NNW that involving a combination of reverse and strike-slip motion. The mechanisms of Seborowski et al. (1982) and Quittmeyer et al. (1985) for earthquakes near Indian Point are of that type. Ratcliffe (1975, 1976) reported a number of brittle faults in the vicinity of Indian Point, including one "small fault with slickensided surfaces found adjacent (immediately

north of) the foundation of reactor 3" that are suitably oriented such that they could be activated in the present stress field.

IV. NRC Staff Responses to Riverkeeper Letter of 2004

The letter from Holden (2004) contains several replies by NRC staff to seismic issues raised earlier in 2004 by Alex Matthiessen, Executive Director, Riverkeeper, Inc. Several of those replies contain much more current information, especially about probabilistic seismic hazard analyses. They state "In response to GL 88-20, Indian Point completed a comprehensive IPEEE review in 1995." That information, however, is not included in the two updated FSARs for Indian Point.

Moreover, several of the responses by the NRC staff to Matthiessen are incorrect. For example, they state "In the area around the Indian Point plant site, there is no evidence to indicate that earthquakes nucleate at unusually shallower depth." To the contrary, depths for several earthquakes near the plants as recorded by the Indian Point network from 1976 to 1983 ranged from as shallow as 1 km to as deep as 12 km (Seborowski et al., 1982; Thurber and Caruso, 1985). Seeber and Dawers (1989) report depths of 4.5 to 5.5 km for aftershocks of the 1985 Ardsley NY earthquake. NRC staff report calculations for a shock of magnitude 5.7 and an historical earthquake of magnitude of 5.2, each at an epicentral distance of 14 km and assumed depth of 10 km. Earthquakes near Indian Point have occurred both closer and shallower than those values.

Another staff response states "It is not possible to determine the rupture lengths of the 1737 and 1884 earthquakes since there are no records to indicate any surface rupture at the time these earthquakes took place." This statement is not correct. It is possible to estimate rupture length from the magnitudes of those events as well as those for the earthquakes of 1783 and 1848 (Fig. 2). This is done routinely by seismologists and earthquake engineers. Few moderate to large earthquakes in the eastern and central North America have involved surface rupture.

V. Earthquake Risk as well as Hazard Need to be Considered in Safety Analyses

Discussions of earthquakes in the updated FSARs consider earthquake *hazard* but not earthquake *risk* to the surrounding regions of high population and assets that could result from

damage to one or more of the reactors. Risk as used here (and by USGS and FEMA) is the product of *hazard* times *people or assets* affected times their *vulnerability*. The area of Figures 2 has a lower earthquake *hazard* compared to those of say California and Nevada, but high seismic *risk* that results from the high vulnerability of its built environment and its very high population (Tantala et al. 2003). New York City, Newark, Trenton, and Philadelphia as well as their surrounding highly-populated surrounding areas are situated in Figure 2. The population of that area was 21.4 million in 2005. FEMA (2001) calculated annualized earthquake losses for 40 large U. S. cities using their program HAZUS. They rank New York City 11th in the nation by that measure of *risk* even though it ranks lower in terms of earthquake *hazard*. It is earthquake *risk* that has increased enormously since Henry Hudson sailed up the Hudson River in 1609. Risk is likely to continue to increase if critical facilities such as Indian Point are not better shielded from earthquake hazards in the greater New York City area.

The shock of 1884 of M 5.25 is the largest known event in Figure 2. The front pages of several New York newspapers for the next day were devoted to that earthquake and the damage it caused. Tantala et al. (2003) used HAZUS with a modified building stock for the Metropolitan New York area to estimate losses for earthquakes of magnitude (Mw) 5, 6 and 7 at the site of the 1884 shock as well as probabilistic calculations for average return periods of 100, 500 and 2500 years. For Mw 6 and 7 events at the site of the 1884 shock they calculate losses from buildings and income of $39 billion and $197 billion respectively. Inclusion of infra-structural losses would about double those figures (K. Jacob, personal communication, 2007). Extrapolated repeat times for the area of Figure 2 for events of M 6 and 7 are about 670 and 3400 years (Sykes and others, 2007). The corresponding probabilities of occurrence in a 50-year period are about 7% and 1.5% respectively. The probability of an earthquake the same size as the 1884 event during a 50-year period is about 22%. Probabilistic hazard assessments, such that in Figure 1, rely on extrapolating rates of earthquake occurrence to time periods longer than historic records.

Probabilistic calculations for Indian Point reactors 2 and 3, such as those used by USGS for their national earthquake hazard maps and those now required by NRC for newer nuclear power reactors, need to be debated and evaluated by wide scientific and policy communities. That approach necessitates the inclusion of rates of earthquake activity for periods longer than the historic record, which was not required under the regulations that existed when the Indian Point reactors were originally licensed. If 20-year license extensions are granted, 60 years of

operation of the two reactors is a sizable fraction of the 270-year historic record of earthquakes. The chance that the reactors could be shaken by intensities greater than VII and/or subjected to accelerations larger than 0.15 g can be calculated and is not negligible.

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New York, Inc., June 30, 19 pp. plus 1 figure (map).

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Figure 1

Eastern portion of earthquake hazard map for the conterminous United States (Frankel and others, 2005) prepared by the United States Geologic Service (USGS) showing 2% probability of exceedance in 50 years for horizontal spectral acceleration with period of 0.2 seconds (frequency of 5 cycles per second=5 Hz.). Acceleration is expressed as percent of gravitational acceleration at surface of earth. Note the "bull's eye" of higher values in New York City, northern New Jersey, and Westchester County, New York, which includes the 3 units at the Indian Point Nuclear Power Station.

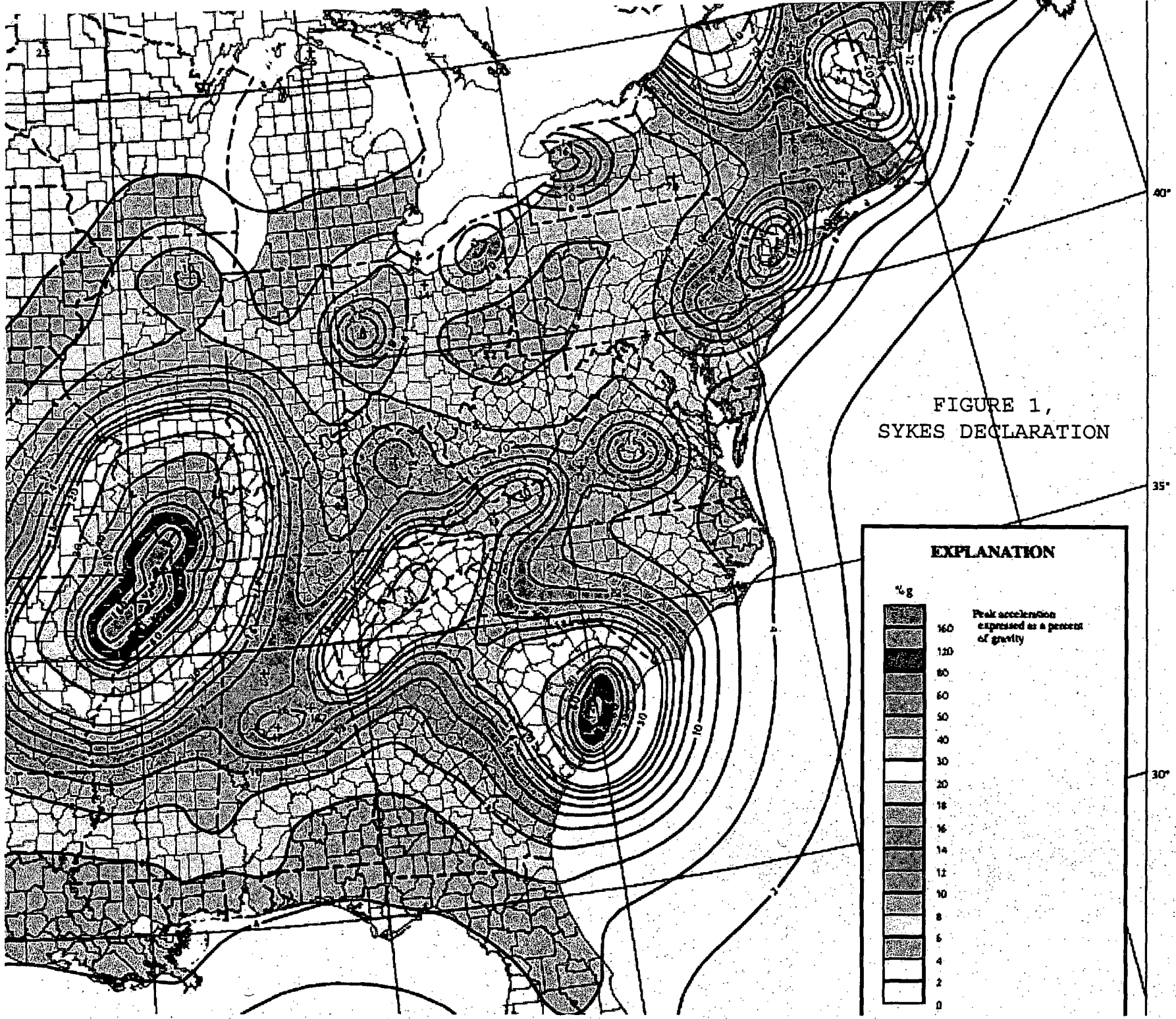


FIGURE 1,
SYKES DECLARATION

Figure 2

Known earthquakes of magnitude ≥ 3.0 in the greater New York City-Philadelphia area from 1677 through 2006 (Sykes et al., 2007).

Large "X" denotes Indian Point Nuclear Power Station. Pink denotes Pre-Cambrian rocks; yellow denotes Mesozoic rocks of Newark basin.

Place names: K = Kingston NY, NYC = New York City, PHIL = Philadelphia, N = Newburgh NY, P = Poughkeepsie NY, SI = Staten Island NY, T = Trenton NJ, Wf = Wappinger Falls NY.

Geological features: BV= Buckingham Valley, CL = Cameron's Line, FF = Flemington-Furlong fault, GPS = Green Pond syncline, HF = Hopewell fault, Hud High. = Hudson Highlands, HVF = Huntingdon Valley fault, Man. Prong = Manhattan Prong, NBB = New York Bight basin, SHB = Sandy Hook basin.

Many additional faults in the Reading Prong and short brittle faults in the Manhattan Prong are not shown. Epicenters of large events of 1737 and 1783 may be uncertain by 100 km and are shown as open circles. No events occurred behind legend. Horizontal projections of P axes of better-determined focal mechanisms of earthquakes and directions of maximum horizontal compressive stress from two sets of hydrofracture and one set of borehole breakout experiments are indicated by inward-pointing arrows. The 1957 and 2003 shocks likely occurred at depth in older rocks of the Reading Prong.

FIGURE 2, SYKES DELCARATION

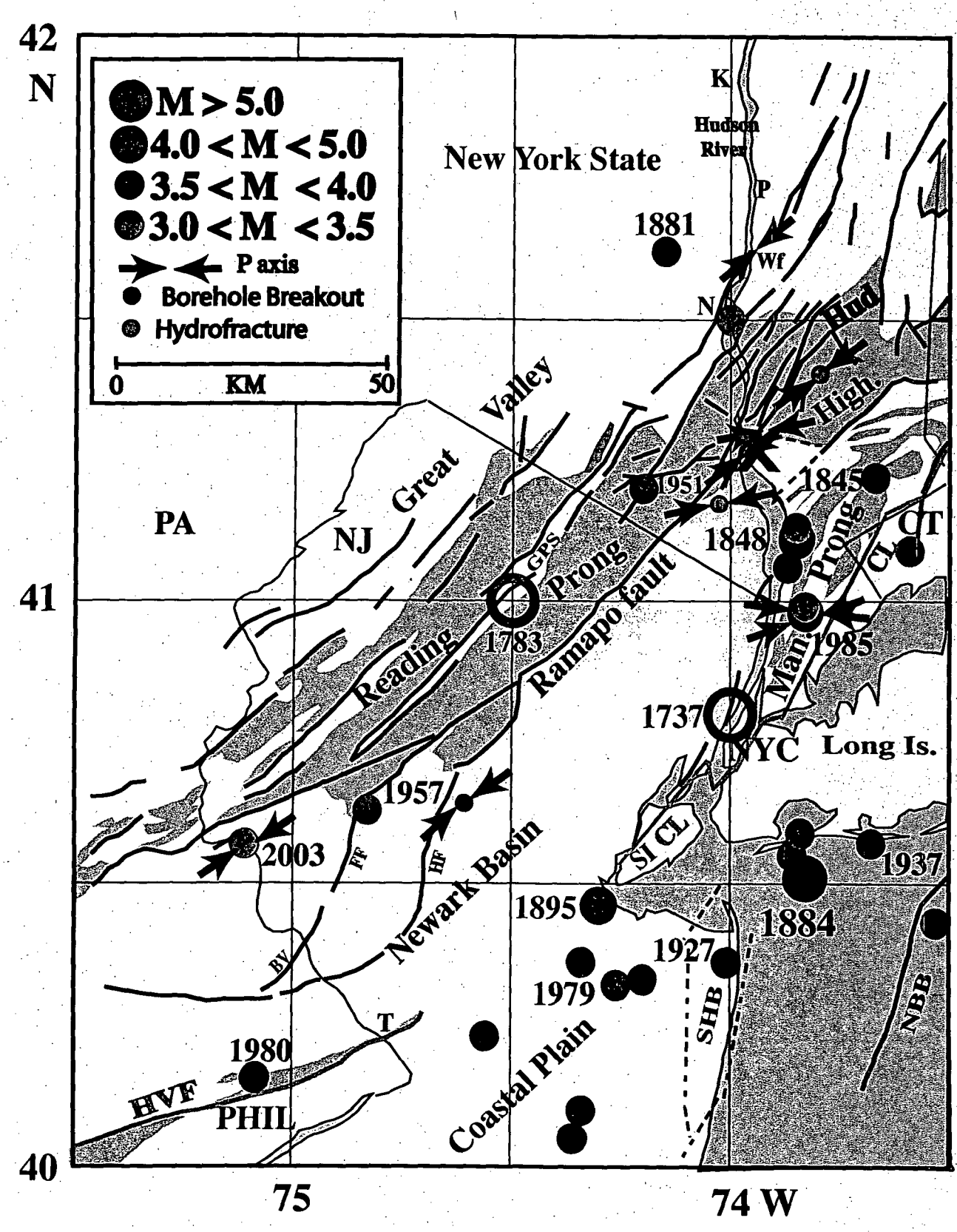


Figure 3

Instrumental locations of earthquakes from 1974 to 2007 (Sykes et al., 2007).

Large "X" denotes Indian Point Nuclear Power Station.

Arrows denote approximate southeastern boundary of Ramapo seismic zone (RSZ) and northwesterly-striking seismic boundary between Stamford CT and Peekskill NY. Pink numerals denote distance along Ramapo zone. Most of the instrumentally-located earthquakes beneath the Newark basin occurred near its northeastern end where its basement shoals.

FIGURE 3, SYKES DECLARATION

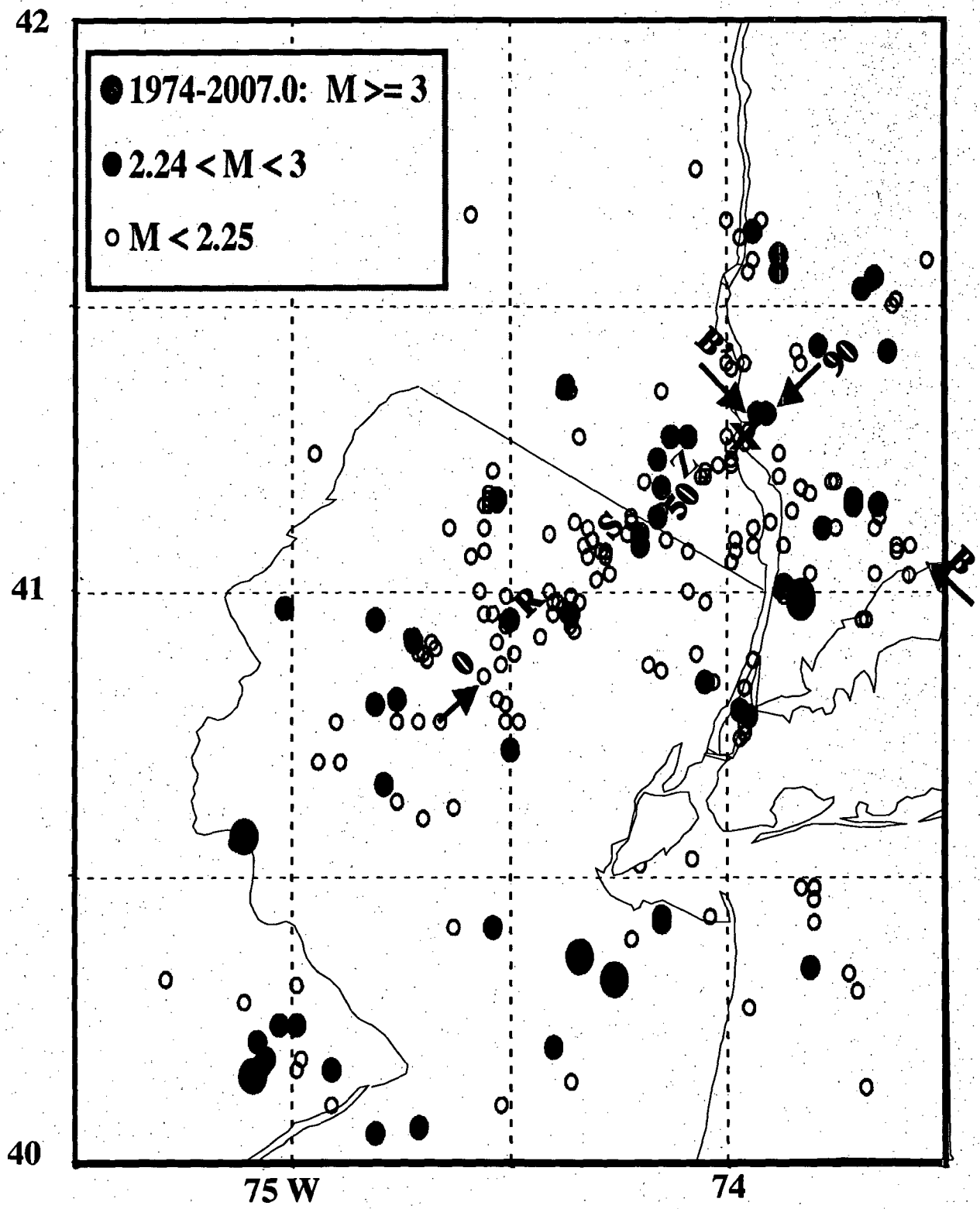


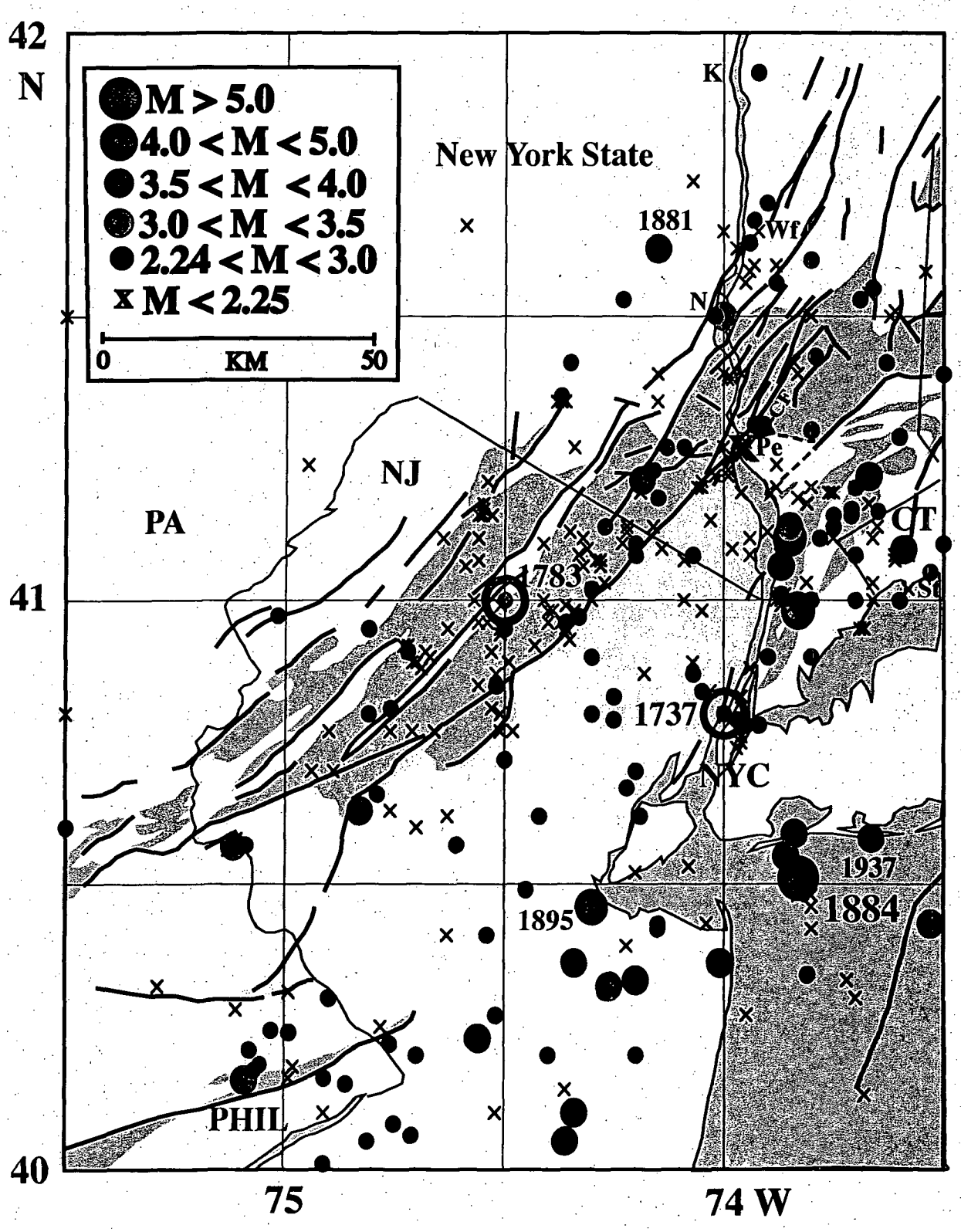
Figure 4

Entire catalog of known earthquakes in greater New York City- Philadelphia area from 1677 through 2004 (Sykes et al., 2007).

Large "X" denotes Indian Point Nuclear Power Station. No events occurred behind legend.

Most of the smaller earthquakes in the Newark basin occurred prior to 1974; many were felt at only a single locality. The population of the basin has long been much higher than that of the Reading Prong-Hudson Highlands. Hence, the record of historic activity in the Newark basin likely over-portrays its rate of microearthquakes activity relative to that of sparsely populated regions. Rock units, faults, magnitudes, open circles and place names same as in Fig. 2. CF = Canopus fault, Wf = Wappinger Falls NY, Pe = Peekskill NY, St = Stamford CT.

FIGURE 4, SYKES DECLARATION



CURRICULUM VITAE

LYNN R. SYKES
April 2007

Present Position

Higgins Professor Emeritus of Earth and Environmental Sciences
Columbia University

Business Address

Lamont-Doherty Earth Observatory of Columbia University
Palisades, New York 10964
Telephone: 845-359-7428, Fax: 845-365-8150
e-mail: sykes@ldeo.columbia.edu

Education

Ph.D., Columbia University in Geology, 1965
Worked under Professor Jack Oliver at Lamont-Doherty Geological Observatory in the field of earthquake seismology.
Dissertation: The propagation of short-period seismic surface waves across oceanic areas.

B.S. and M.S. degrees both awarded by Massachusetts Institute of Technology, Cambridge, MA, Department of Geology and Geophysics, with emphasis on geophysics.

Other areas of concentration: Physics, mathematics, electrical engineering.

Scholarships, Fellowships, Awards, and Honors

Proctor and Gamble Scholarship, M.I.T. for four years as an undergraduate.

Edward John Noble Leadership Award during first three years of graduate study.

Summer Research Fellow, 1959, Woods Hole Oceanographic Institution, Woods Hole, MA.

Fellow of American Geophysical Union, Geological Society of America, Royal Astronomical Society, American Association for Advancement of Science, Geological Society (London).

H.O. Wood Award in seismology from Carnegie Institution of Washington for research in geologic aspects of seismology: $10,000, 1967-1970.

Associate Editor, Journal of Geophysical Research, 1968-1970.

Sloan Fellow, 1969-1971.

Presented Frontiers in Geophysics paper "The New Global Tectonics" before 50th Anniversary Meeting of American Geophysical Union, April 1969.

President of Geological Section, New York Academy of Sciences, 1970-1971.

Macelwane Award to outstanding young geophysicist for 1970 by American Geophysical Union.

President of Section on Tectonophysics of American Geophysical Union, 1972-1974.

Walter H. Bucher Medal of the American Geophysical Union for "original contribution to the basic knowledge of the earth's crust", 1975.

Elected to U.S. National Academy of Sciences, 1979.

Chosen as Higgins Professor of Geology, Columbia University, 1978.

Work on Plate Tectonics included in the exhibit, "Creativity -- The Human Resource", California Academy of Sciences, San Francisco, April-May 1979.

Elected to American Academy of Arts and Sciences, 1979.

Sherman Fairchild Distinguished Scholar, California Institute of Technology, 1981.

Visiting Fellow, Clare Hall, Cambridge University, Spring 1982.

President-Elect of Section on Seismology, American Geophysical Union, 1982-1984; President 1984-1986.

Public Service Award for 1986 from Federation of American Scientists (along with Jack F. Evernden and Charles Archambeau) for "leadership, effectiveness, and courage in the application of seismology to the banning of nuclear tests, through public education and bureaucratic struggles".

Honorary Doctoral Degree by the State University of New York at Potsdam, May 1988.

Fellowship, John Simon Guggenheim Memorial Foundation, October 1, 1988-June 30, 1989.

John Wesley Powell Award by the U.S. Geological Survey for service to U.S. earthquake program, May 1991.

Finalist, Mayor of New York City's Award for Excellence in Mathematical, Physical and Engineering Sciences, 1992.

Medal of Seismological Society of America, March 1998.

Vetlesen Prize, for "achievement in the sciences resulting in a clearer understanding of the earth, its history or relation to the universe" January 24, 2000

Scientific Research

Scientific research includes investigations of long-period and mantle seismic waves (1961-1962); surface wave propagation across ocean areas (1961-1964); precise location of earthquake hypocenters and relationship of spatial distribution of earthquakes to large-scale tectonic phenomena (1962-); seismicity of island arcs, mid-ocean ridges, and fracture zones (1963-); field study of aftershocks of 1964 Alaskan earthquake (1964); field study of deep and shallow earthquakes in Fiji-Tonga region (1966); spatial and temporal distribution of major earthquakes and major aftershock series (1965-); field study of microearthquakes in Iceland (1968); earthquake prediction (1967-); seismology and the new global tectonics (1968-1975); field study of Denali fault, Alaska (1967); field study of microearthquakes in Nevada and in Puerto Rico - Virgin Islands region (1969); discrimination between earthquakes and underground explosions and implications for a nuclear test ban treaty (1962-); seismicity and the tectonics of eastern North America (1969-1985); state of stress in the interiors of plates and intraplate earthquakes (1969-1980, 1994-); seismicity, tectonics and earthquake prediction in Puerto Rico and Virgin Islands (1974-1984); spatial and temporal variations of seismicity in California (1981-); nuclear arms control (1970-); seismic safety of nuclear power plants and their spent fuel pools (2004-).

Employment

Present Status: Higgins Professor Emeritus of Earth and Environmental Sciences, Columbia University.

Head of Seismology Group (1973-1983)

Earth Sciences Laboratories, Environmental Science Services Administration, Department of Commerce as Research Geophysicist; Adjunct Professor of Geology, Columbia University, June 1966-August 1968. GS 14.

Lamont-Doherty Geological Observatory of Columbia University; Research Associate in Seismology, 1964-1966. Research Assistant, 1961-1964.

Woods Hole Oceanographic Institution, Woods Hole, MA, Summers of 1959 and 1960, research in marine geophysics, included participation in scientific cruises

in Atlantic and Mediterranean. Master's thesis in conjunction with M.I.T. -- "Correlation of Physical Properties of Deep-Sea Sediments with Sea-Bottom Reflections."

U.S. Geological Survey, Geophysical Laboratory, Silver Spring, MD, summer of 1956 as physical science aide. Studied consolidation of calcium carbonate muds with E.C. Robertson and M. Newell.

Professional Societies

Seismological Society of America, American Geophysical Union, Royal Astronomical Society, Geological Society of America, New York Academy of Sciences, Geological Society of London, U.S. National Academy of Sciences, American Academy of Arts and Sciences, Arms Control Association, Federation of American Scientists. Originated ideas of forming Southern California Earthquake Center and Alaskan Volcano Center.

Scientific Committees and Advisory Boards

Polar Geophysics Panel of National Academy of Science (1968).

Advisory Committee of National Academy of Sciences to ESSA Research Laboratories on solid earth geophysics (1968-1969).

NASA Geodesy and Cartography Subcommittee of Space Science and Applications Steering Committee (1968-1970).

NAS/NRC Committee on World-Wide Standardized Seismograph Network (1969).

Organizing Secretary: International Symposium on Mechanical Properties and Processes of the Mantle, sponsored by International Upper Mantle Committee (1970).

Member Board of Directors of Seismological Society of America (1968-1972).

Member JOIDES panel on Deep Crustal Drilling in Marine Areas (1970-1971).

Member of U.S. Geodynamics Panel on Mid-Atlantic Ridge (1971-1972).

Committee on Seismology of National Academy of Sciences/National Research Council (1972-1975).

Testified before U.S. Senate Foreign Relations Committee, Subcommittee on Arms Control, International Law and Organization on Hearings on Comprehensive Test Ban Treaty, May 15, 1972.

Advisor to New York State Geological Survey and New York State Environmental Protection Agency on Earth Hazards Related to Fluid Injection (1970-1974).

Chairman of Search Committee for Director of Lamont-Doherty Geological Observatory, 1972.

Member of Vetlesen Award Committee, Columbia University (1972-1985).

Member of Executive Committee (1973-1977) and Advisory Board (1973-1981; 1999-) Lamont-Doherty Geological Observatory of Columbia University. Chairman of Advisory Board (1975-1981).

Panel on Earthquake Prediction, National Academy of Sciences/National Research Council (1973-1976).

Advisory Committee on Proposals for Earthquake Prediction, U.S. Geological Survey (1974).

Member of Working Group, U.S./U.S.S.R. Joint Program for Earthquake Prediction (1973-1978).

Member U.S. Technical Delegation for talks on treaty on Threshold Limitations of Underground Nuclear Explosions, Moscow, June-July 1974.

National Science Foundation Earth Science's Review Panel (1974-1977).

U.S. Geodynamics Committee, study groups on plate interiors and Cocos and Caribbean plates.

Member of U.S. Delegation on Earthquake Prediction during visit to U.S.S.R., October 1973.

Member of U.S. Seismology Group for visit to People's Republic of China, October - November, 1974.

Visiting Professor, Earthquake Research Institute of Tokyo University, November-December 1974.

Nominations Committee on Fellows for American Geophysical Union, 1975, and Committee on Publications 1975-1976. Committee for Bucher Medal, 1977, Committee for Bowie Medal, 1986-87, Chair, Bucher Medal, 2001.

Consultant and expert witness for New York State and for Citizen's Committee for the Protection of the Environment involving seismologic and geologic safety of Indian Point Nuclear Power Reactors, New York, 1975-1976.

Review committee for U.S. Federal Government program on Earthquake Prediction and Hazards Reduction under President's Science Advisor, Dr. H. Guyford Stever (1976).

Planning committee for Chapman Conference on "State of Stress in the Lithosphere", Aspen, Colorado, 1976.

U.S. Air Force -- AFTAC -- panel on seismic determination of yield of underground nuclear explosions, 1974-1977.

Defense Science Board -- panel on yields of underground nuclear explosions, 1977.

Committee on award of Day Medal -- Geological Society of America, 1974-1976; Penrose Medal, 1977-1978.

Member Seismological Panel, Office of Science & Technology Policy, Executive Office of the President, September 1 & 2, 1977.

Member USGS Earthquake Studies Advisory Panel, 1977-1981.

Member Columbia University Commission on Academic Priorities in the Arts and Sciences, 1978-1979.

Testified about Seismic Risk to Indian Point, NY, Nuclear Power Plants before Advisory Committee on Reactor Safety, U.S. Nuclear Regulatory Commission, June 16, 1978.

Member Search Committee for Director of New York State Geological Survey, 1978.

Chairman, Panel of Public Policy Regarding Prediction of Earthquakes, American Geophysical Union, 1979-1980.

Lecturer, NATO Summer Institute on "State of Stress in the Earth's Lithosphere", June 1979.

Convenor, Ewing Symposium on Earthquake Prediction, May 12-16, 1980.

Member Earthquake Prediction Evaluation Council, U.S. Geological Survey, 1979-1982.

Chairman of Search Committee for Director of Lamont-Doherty Geological Observatory, 1981.

Lecturer, NATO Summer School on Earthquake Risk, Guadeloupe, August 1983.

Member Advisory Panel on Seismology, Defense Advanced Research Projects Agency, 1983-1988.

Co-organizer of symposium "Verification of Nuclear Test Ban Treaties", with Dr. Jack Evernden, American Geophysical Union, Baltimore, MD, June 1983.

Co-organizer of interdisciplinary course for undergraduates "The Nuclear Arms Race", Columbia College, 1984, 1985.

Member, Columbia University Seminar on Arms Control, 1984-1996.

Chairman, National Earthquake Prediction Evaluation Council (NEPEC), U.S. Geological Survey, Fall 1984-Summer 1988.

Participant in NOVA television program "Spacebridge to Moscow", October 2, 1984.

Member of Expert Review Committee for Evaluation of National Earthquake Hazards Reduction Program, 1987.

Member of the Advisory Panel to Natural Resources Defense Council on Seismic Stations in the Soviet Union, 1986-1988.

Member of Seismic Verification Advisory Panel, Office of Technology Assessment, U. S. Congress, 1986-1987.

Seismology Seminar Course at Columbia University on Verification of Nuclear Test Ban Issues, with Paul G. Richards, Fall 1987 semester.

Participant, Belmont Conference on Nuclear Test Ban Policy, Fall 1988.

Member of U.S. National Committee for the Decade for Natural Hazards Reduction, National Academy of Sciences, National Reseaarch Council, 1989-1990.

Participant in NOVA program "Earthquake", 1990.

Invited Speaker, Princeton Symposium on Non-Proliferation and Nuclear Testing, Nov. 1992.

Co-Organizer, All Union Symposium on Verification of Treaties to Limit the Testing and Proliferation of Nuclear Weapons, American Geophysical Union Spring Meeting, May 27, 1993.

Invited talk "Earthquake Prediction", regional meeting of National Academy of Sciences, at Columbia University, Fall 1996.

Invited talk "Earthquake Prediction: What is Possible and what is Unknowable", Columbia Graduate students in science, April 9, 1997.

Co-organizer, symposium on "Earthquake Stress Triggers, Stress Shadows, and their Impact on Seismic Hazard", Menlo Park, CA, March 21-22, 1997.

Co-organizer, symposium on "Earth Systems Predictability: The Unknown and Unknowable", Santa Fe Institute Workshop, November 6-8, 1997.

Head, Natural Hazards Initiative, Lamont-Doherty Earth Observatory, 1998-99

Co-Organizer, All Union Symposium on Verification of Comprehensive Nuclear Test Ban Treaty, American Geophysical Union, Spring Meeting, May 31, 2000.

Testified in 2000 before Arms Control Advisory Board, U. S. State Dept., Jason group, Stanford-Lawyers Alliance for World Security, National Academy of Sciences/National Research Council Committee on Comprehensive Nuclear Test Ban Treaty.

Board of Directors, Federation of American Scientists, 2000-2003; also on their panel on CTBT.

American Geophysical Union, Chair Bucher Award, Fall 2001

PUBLICATIONS

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1964 Sykes, L.R., Deep-focus earthquakes in the New Hebrides region, J. Geophys. Res., 69, 5353-5355.

1965 Sykes, L.R., The seismicity of the Arctic, Bull. Seismol. Soc. Am., 55, 536-591.

1965 Sykes, L.R., and M. Ewing, The seismicity of Caribbean region, J. Geophys. Res., 70, 5065-5074.

1966 Tobin, D.G., and L. R. Sykes, Relationship of hypocenters of earthquakes to the geology of Alaska, J. Geophys. Res., 71, 1659-1667.

1966 Sykes, L.R., Spatial and temporal distribution of earthquakes, in ESSA Symposium on Earthquake Prediction, pp. 38-43, U.S. Dept. of Commerce, Washington, D.C.

1966 Sykes, L.R., Seismicity and deep structure of island arcs, J. Geophys. Res., 71, 2981-3006.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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In re:

License Renewal Application Submitted by

Entergy Nuclear Indian Point 2, LLC,
Entergy Nuclear Indian Point 3, LLC, and
Entergy Nuclear Operations, Inc.

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Docket Nos. 50-247-LR, 50-286-LR

ASLBP No. 07-858-03-LR-BD01

DPR-26, DPR-64

DECLARATION OF LEONARDO SEEBER

Leonardo Seeber hereby declares under penalty of perjury that the following is true and correct:

1. I am currently a senior research scientist at the Lamont-Doherty Earth Observatory of Columbia University. I have served as a research scientist at Lamont Doherty since 1972. I received a B.S. in Nuclear Engineering from Columbia University in 1965.

2. During the course of my career, I have studied earthquake related issues in the New York City Seismic Zone (which includes portions of New York State, New Jersey, Pennsylvania, and Connecticut) and throughout the United States and the world. My CV is attached to this declaration.

3. I have prepared a report concerning the earthquake activity in intraplate continental regions such as eastern North America, with emphasis on issues directly relevant to earthquake hazard in the greater tri-state New York City Seismic Zone and the area in and around the site for the Indian Point Nuclear Power Station.

4. The report and CV are true and correct to the best of my personal knowledge.

5. Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Dated:

November 29, 2007
Palisades, New York

Leonardo Seeber

Leonardo Seeber presents the following report in connection with the application to renew the operating licenses for the Indian Point Nuclear Power Station for an additional 20 years.

Premise

These comments summarily characterize earthquake activity in intraplate continental regions such as eastern North America, with emphasis on issues directly relevant to earthquake hazard in the greater tri-state New York City seismic zone (NYCSZ) and at the Indian Point site. The last three decades have witnessed substantial improvements in knowledge about earthquakes and their relation to geologic features. This progress includes generalities about intraplate areas, where earthquake activity and geologic processes are relatively subdued, as well as specifics about the NYCSZ. Particularly relevant are differences in understanding between the present and the time Indian Point 3 was licensed. Some of these differences would unambiguously increase the estimated hazard at Indian Point.

In New York City seismic zone, which includes the site for the Indian Point Nuclear Power Station, some of the earthquakes occur at mid-crustal depths, but many reliable earthquake hypocenters show that very shallow earthquakes predominate in the New York City Seismic Zone including the area around Indian Point.

Furthermore, many of the larger earthquakes in the NYCSZ during the last three decades have occurred on northwest-striking faults. These faults are small relative to the Ramapo and other northeast-striking faults, but form a distinct family of faults that generally cut and are therefore younger than other faults. One of these faults has been mapped 5 km north of Indian Point and maybe related to a nearby earthquake sequence that occurred during 1977-1980.

Generally, earthquake activity and tectonics along active plate-boundaries and in intraplate zones were thought to differ in rate, but otherwise to stem from the same fundamental process and thus to resemble one another qualitatively. Improved observations and understanding has revealed distinctions in both the spatiotemporal distribution of earthquakes and their source characteristics. These differences affect algorithms used to derive ground motion from earthquake parameters and thus have important implications for hazard. The scope of these comments, however, is strictly

qualitative. It does not include any evaluation of the hazard, nor a sensitivity analysis to evaluate the effects on the hazard of specific aspects of intraplate earthquake generation, as they are now understood.

Intraplate Eastern North America and The New York City Seismic Zone

Eastern North America is a continental area fully contained within the North American plate. This intraplate area is one of the more thoroughly studied from geological and seismological viewpoints. During the historic period earthquakes have occurred over most of this area, although epicenters are more concentrated in specific zones. Widespread earthquake activity is consistent with stress measurements, which show that the stress in the upper crust of intraplate continental areas is generally high and near failure (e.g., Zoback and Zoback, 1989). Each earthquake manifests fault rupture and slip and thus contributes to deformation of the Earth's crust. Deformation events associated with earthquakes are generally coherent over eastern North America and cause shortening in an east-northeast direction (Sbar and Sykes, 1973). The rate of this deformation has been calculated from the earthquake activity (Anderson, 1986). This rate is slower than can be resolved with current geodetic measurements but is significant when applied over geologic time scales. The accumulated effects of this intraplate strain have been associated with widespread geologic structures (e.g., Sbar and Sykes, 1973), but these structures are generally subtle and rarely include geologic evidence of fault slip that could be associated with specific intraplate earthquakes, such as the surface rupture associated with 1990 earthquake in Ungava, northern Quebec (Adams et al. 1991). Thus geologists have described eastern North America and similar intraplate areas as "stable continental regions" (SCR). The combination of widespread intraplate earthquake activity and lack of obvious geologic strain is still generally astonishing and is the subject of ongoing research (further discussed below). This geologic 'stability', however, should not mislead regarding intraplate earthquake activity, which is observed, has caused damage, and will likely continue to do so. Geologic stability was clearly a factor a few decades ago in the debate about earthquake hazard at Indian Point. The apparent geologic stability in the area around Indian Point was contrasted with the obvious geologic activity along plate boundaries such as the San Andreas fault zone in California. More useful is a comparison with other intraplate areas that display similar 'stability', yet have experienced

large and destructive earthquakes (e.g., Coppersmith and Joungs, 1989; Jonston, 1989).

The greater New York City metropolitan area correlates with a seismic zone (NYCSZ), a concentration of earthquake activity that stands out in the field of epicenters over eastern North America. Geologically, the NYCSZ is associated with the Newark Basin (Figure 2), a feature formed during the opening of the Atlantic Ocean in the Mesozoic era. Earthquakes are concentrated in older rocks that outcrop around the basin, from Reading, Lancaster, and Philadelphia PA, to Peakskill NY and from Westchester and New York City to the Hudson Highlands and the lower Hudson Valley. The NYCSZ is one of several zones of concentrated earthquake activity in eastern North America that have persisted through the historic period, and have been documented by both early felt and damage reports as well as by current instrumental data (e.g., Hough et al., 2003). Some of these zones have experienced very large earthquakes ($M \geq 7$) that caused damage, such as the 1886 Charleston SC earthquake, or would have caused much more damage had they occurred later, such as the 1811-12 New Madrid MO earthquakes. Other earthquake zones, such as the ones in eastern Tennessee and the NYCSZ have generated intermediate-size earthquakes with relatively minor consequences during the historic period, but are thought to be capable of producing larger earthquakes. The absence from the record of the largest possible earthquakes in these areas is accounted for by average recurrence times likely to be substantially longer than the historic period (e.g., Seeber and Armbruster. 1991).

Relation of intraplate earthquakes to observable faults

In tectonically active regions, such as the San Andreas plate boundary in California, damaging earthquakes occur on faults that can be independently recognized as being active from their displacement characteristics. Slip rate can often be measured directly along the surface trace of faults or assessed indirectly by the rate of growth of fault-related folds. Thus geologic data on faults and their slip behavior can be used to assess the earthquake potential in these areas independently of earthquake data. A fault in California that displays evidence of no geologically recent displacement, during the last tens to hundreds of thousand years, is usually insignificant for earthquake hazard. Early attempts at characterizing intraplate earthquake hazard applied this geologic approach seeking to identify key faults responsible for most of

the earthquakes. Past rupture behavior of these faults would then offer strong geologic constraints on future earthquake activity (e.g., Crone et al., 1992). Many geologic studies of the source areas following intraplate earthquakes, however, have dashed these hopes (e.g., NRC Regulatory Guide 1.208, p. 6, 2007).

Most of the large intraplate earthquakes (M≥6) worldwide are in the shallow part of the crust (Figure 1). Not surprisingly, these large shallow earthquakes tend to rupture the surface (Table 1), thus unequivocally identifying the causative fault and offering opportunities for geologic studies of these faults. After they ruptured in M≥6 earthquakes, these faults are declared active. Had those fault been studied before the earthquakes, however, they would probably not have been considered active, because they often showed no sign of having ruptured during the previous hundred thousand years or much longer (e.g., Crone et al., 1992; Machette et al., 1993; Adams et al., 1991; Seeber et al., 1996). Accumulated displacements on some of the faults that produced significant earthquakes in the NYCSZ were remarkably small and allowed for no more than a few surface-rupturing earthquakes during prior time those faults existed in an intraplate regime (e.g., Seeber and Dawers, 1989; Dawers and Seeber, 1991; Seeber et al., 1998). Multiple prehistoric surface ruptures closely spaced in time were discovered on some of the intraplate faults (e.g., the Meers fault in Oklahoma; Crone and Wheeler, 2000). They seem to be clusters of events, which are nevertheless preceded by long periods of quiescence. Long quiescence, therefore, seems to characterize intraplate faults known to have produced damaging or potentially damaging earthquakes.

The rate at which individual "active" intraplate faults produce earthquakes is low even in comparison to the overall rate of earthquake activity in these areas. This has lead to the hypothesis that intraplate deformation and earthquakes are distributed among many faults, including minor ones (e.g., Seeber et al., 1986). The contribution from each fault is small, but together they account for the earthquake activity and hazard in intraplate areas. In order to produce very large earthquakes (M≥7), some of these faults are large, possibly having played a major role in a previous more active geologic regime. Even these large faults however, do not seem to have accumulated much strain in the current intraplate regime. The 1811-12 New Madrid sequence in the central US and the 1819 and 2001 earthquakes near Bhuj, western India, are among the larges known intraplate earthquakes. Both sequences include reverse faulting that thickens the crust and thus tends to

increase topography. Yet these earthquakes have occurred in parts of the continents barely above sea level and with no evidence of sustained uplift.

Along plate boundaries, few master faults deliver much of the strain release and hazard. This concentration of strain is consistent with strain-softening, which seems to control many geological phenomena, including the reactivation of pre-existing faults widely noticed for intraplate earthquakes (e.g., Sykes 1978). The partitioning of strain over many faults is counter to strain-softening behavior and may be symptomatic of widespread application of the forces that drive intraplate deformation. In any case, the important conclusion for hazard is a negative one: geologic evidence of no rupture in recent geologic time (e.g., $\geq$0.1 million years) and/or very slow slip rates are not useful criteria to characterize the earthquake potential of an intraplate fault. The concept of 'capable fault' was developed in active areas. Given current understanding about intraplate earthquake and deformation regimes, geologic information needs to be applied differently in the evaluation of earthquake hazard in those areas. Rather than time since last rupture or slip rate, more useful criteria would be fault geometry, size, and orientation relative to current stress regime. Some of the faults shown to generate earthquakes in the NYCSZ are secondary, but appear to share a common geometry – a NW strike – and to be genetically related. Another important role of geology is therefore to identify families of faults that are likely to behave similarly in the current intraplate regime.

Given that intraplate earthquakes are distributed on many faults, should all faults be considered potential sources of earthquakes? No. Evidence often points to specific faults or families of faults and the NYCSZ offers a case in point. The NYCSZ is centered along the Appalachian thrust-fold belt. This 'compressional' belt developed during a long-standing convergence boundary and repeated plate collisions during the Paleozoic era. During the Mesozoic, the era of dinosaurs, this belt of shortening and mountain building became a rift zone of extension and developed a series of basins by reversing motion on some of the faults active during horizontal shortening. The Ramapo fault offers a prominent example of this behavior (e.g., Ratcliffe et al., 1986). From this long tectonic evolution, the eastern seaboard of North America inherited a set of major NE striking faults that control the exposed lithology and thus the current morphology of the Appalachian Mountains. The inherited structures include also secondary faults, which are generally smaller in both lateral dimensions and accumulated displacement. Prominent among the secondary faults in the NYCSZ, is a set striking NW,

approximately perpendicular to the northeasterly strike of the Appalachians (Figure 2; Hall, 1991; Dawers and Seeber, 1991).

The Ramapo Fault

The fault system that borders the Newark Basin, including the Ramapo fault, is one of the most prominent structural and geomorphic features within the NYCSZ. This fault system played a critical role in both the compressional phase that formed the Appalachians and in the rifting that followed (e.g., Ratcliffe et al., 1986). It is now associated with the most prominent feature in the spatial distribution of earthquakes of the NYCSZ. On a map view, epicenters are concentrated along the Hudson Highlands and increase in density to the SE reaching a maximum below the trace of the Ramapo fault (Figure 2). In the Newark Basin southeast of the fault earthquake activity is low. Thus the trace of the Ramapo fault marks both a maximum and a boundary in the earthquake activity. This strong spatial association became clear decades ago and was a central issue during early discussions about earthquake hazard at Indian Point, which is very close to the fault. Improved and more abundant earthquake data have since raised two issues. First, these data illuminate the ruptures of several earthquakes in the NYCSZ, some very close to the Ramapo fault. None of these ruptures were on that fault, however, nor were they on similar first-order faults parallel to the Appalachians (e.g., Seeber et al., 1998). Second, the Ramapo fault dips to the SE below the Newark Basin, but the earthquake activity is clustered in a sub-vertical zone below the surface trace of the fault rather than along the subsurface part of the fault (e.g., report by Lynn Sykes). These two observations detract from a simple and straightforward interpretation of the current role of the Ramapo fault and its potential to generate damaging earthquakes. Nevertheless, the obvious spatial correlation between this fault and earthquake activity gives it a central role for earthquake hazard, even if only on statistical grounds. Furthermore, a spatial correlation coupled with lack of earthquakes on the fault itself characterize portions of the San Andreas fault and other master faults during interseismic periods between major ruptures and their aftershocks. The current analogous situation along the Ramapo fault, therefore, does not exclude future earthquakes on this fault.

NW-Striking Fault Set

Most of the secondary NW-striking faults in the NYCSZ have subtle structural and geomorphic surface expressions. Nevertheless, a number of them have been traced over many kilometers and have been named and mapped (Hall, 1991, Dawers and Seeber, 1991; Seeber and Dawers, 1989; Mergurien, 1986; Baskerville, 1982). The 125th street fault across northern Manhattan is one of the better developed and better known of these faults. Geologic data from surface outcrops and in tunnels indicate a mixture of normal and strike slip faulting thought to pertain to a phase of rifting during the Mesozoic. Pre-existence of at least some of these faults during Paleozoic contraction is probable. Despite the relatively minor role of these faults in accommodating strain in the previous tectonic regime, they seem to play a major role in the current intraplate regime. Seismological field investigations of a number of recent earthquakes and aftershocks in the NYCSZ have revealed reliable details about the geometry, size and slip orientation of their fault ruptures. In all cases these ruptures strike NW (Pomeroy et al., 1976; Seberowski t al., 1982; Armbruster and Seeber, 1987; Seeber and Dawers 1989; Dawers and Seeber, 1991; Hough and Seeber 1991; Seeber et al, 1998). The 1985 Ardsley earthquake clearly ruptured the Dobbs Ferry fault that had been recognized as one of the NW-striking faults before the earthquake (Figure 2; Hall, 1991; preliminary map widely available before 1985). Some of the other earthquakes have been tentatively associated with known NW-striking faults.

In 1977 to 1980, a sequence of earthquakes was centered 5km NNE of Indian Point. This sequence of earthquakes is thought to have ruptured one or more NW striking fault(s). No mapped faults have been directly associated with the sequence, but one of the two possible fault planes determined from the seismicity would outcrop approximately along the northwest-striking segment of the Hudson River in the Hudson Gorge, just north of Indian Point. This portion of the river is thought to be controlled by a NW-striking fault. A portion of this fault is mapped across the Hudson Highlands NW of the gorge (Figure 2; Seberowski et al., 1982). Another NW-striking fault – possibly the same fault – has been mapped on the east bank of the river 5km NNW of Indian Point (Ratcliffe, 1980). This is a small fault, but is thought to be the youngest of the mapped faults in that area. The existence of such a young fault, which may be related to a sequence of earthquakes, needs to be acknowledged and further examined especially given its proximity to Indian Point.

In addition, the overall pattern of earthquake activity in the NYCSZ is characterized by a sharp NW-striking boundary separating the seismic zone from an aseismic area to the NE. This boundary is well expressed across Westchester, reaching the Hudson Gorge slightly north of Indian Point. This feature in the current seismicity traverses the Manhattan Prong, a geologic terrane characterized by intense deformation and temperature effects interpreted to represent the core of the Appalachian belt. The boundary to the earthquake activity has not been associated with a particular geologic structure or lithologic boundary, except that it is sub-parallel to the NW-striking fault set and may coincide with one or more faults in this set yet to be mapped (Hall 1991). This correlation is comparable with the Ramapo fault serving as the SE boundary of the earthquake activity in the Hudson Highlands. In both cases the significance of the spatial correlation between earthquake activity and these geologic structures is unclear, but it indicates a role of these structures in the current regime and, in the case of the NW-striking fault set, it reinforces the suggestion from studies of individual earthquakes that faults in this set should be considered possible sources of significant earthquakes.

Depth range of intraplate earthquakes

Along the San Andreas transform in California and in many other tectonically active regions, most hypocenters are deeper than 5 km and large earthquakes tend to nucleate near the brittle-ductile transition at mid-crustal depths (10-15km). These large earthquakes may rupture to the surface, yet most of the seismic energy is released in the deeper part of the rupture where high confining pressure keeps the rock strong. Most intraplate areas exhibit a binomial depth-distribution of earthquake activity, markedly different from the distribution in active regions (Figure 1). Many intraplate earthquakes, including large ones, are very shallow. Most of the recent and well-studied $M \geq 6$ intraplate earthquake ruptures are confined within the upper ~5 km of the crust and reach the surface, although the recent very large intraplate earthquake in Bhuj (west India) 2001 ruptured the deep crust and did not reach the surface. Small intraplate earthquakes are also mostly shallow, but difficulty of distinguishing depths in the upper 10 km when stations are sparse leads to a tendency to overestimating depth in routine analyses. Some intraplate earthquakes originate in the mid to deep crust, deeper than 15-20km, a depth range where deformation occurs a-seismically along the San Andreas Fault in California.

The deeper intraplate earthquakes seem to occur primarily along ancient rift zones, no more active as such, but still characterized by large deep-rooted faults. Passive continental margins are in this category. These areas have also been associated with the largest of the intraplate earthquakes, such as the 1811-12 series in New Madrid MO, the 1886 Charleston event, and the one in Bhuj in 2001. The NYCSZ is situated along a passive continental margin and includes prominent rift structures such as the Newark Basin and the Ramapo fault (Figure 2). In this seismic zone, some of the earthquakes occur at mid-crustal depths, but many reliable hypocenters show that very shallow earthquakes predominate. Among these, the Mw4.7 1994 earthquake near Reading is the largest, and ruptured a fault in the upper 2.5 km (Seeber et al., 1998). This earthquake was triggered by a very small stress change caused by a quarry. Nevertheless, this depth is characteristic of natural earthquakes along the Reading Prong. Two M4 earthquakes on the southeast side of the prong, in Lancaster PA 1984 (Armbruster and Seeber, 1987) and Ardsley NY 1985 (Hough and Seeber, 1991), were 4-to-5 km deep. The Ansville sequence of earthquakes, 1977-1980, is centered only 5 km NNE of Indian Point and is centered at a depth of about 2 km (Seberowski et al., 1982).

The shallow depth of many intraplate earthquakes is now gradually being recognized. A few decades ago only small earthquakes insignificant for hazard ("microearthquakes") were thought to originate in the very shallow part of the crust (depth <5km). The same earthquakes raised from a typical California depth range to one appropriate for an intraplate area such as the Reading Prong, would be closer to people and structures and probably cause higher intensity, even without accounting for differences in seismic attenuation. The most obvious effect is to lower the magnitude threshold for which damage can start. This can dramatically increase the number of damaging earthquakes because the number of earthquakes in a given magnitude range increases logarithmically with decreasing magnitude. Thus, if the threshold were to decrease from M5 to M4, the number of damaging earthquakes may increase nearly 10 times. At close distances, these small shallow earthquakes can be expected to produce a burst of shaking of short duration shifted toward frequencies higher than would be expected from larger and deeper events also at the damage threshold. The kind of damage would differ; acceleration would be higher from the shallow events and velocity would be higher from the deeper events. Stress-drop, which affects the source spectrum, and the depth distribution of seismic attenuation, which

affects the intensity fall off with distance, are also likely to play a major role in determining the hazard from shallow intraplate earthquakes. We have recognized that an important component of intraplate earthquake activity is very shallow (Close an Seeber, 2007; Figure 1), but have so far done little to explore how it may differ in other ways from deeper intraplate earthquake activity and how these differences may affect the hazard.

Anthropogenic Earthquakes

Finally, the potential of large intraplate earthquakes to nucleate at shallow depths increases the potential for engineering activities to trigger damaging earthquakes. Stress and pore pressure changes induced by a variety of operations, such as mining, quarrying, and fluid waste disposal, can be significant in the very shallow crust, but rarely at depth where large California earthquakes tend to nucleate. Thus anthropogenic earthquake activity has been a minor concern along the San Andreas plate boundary and in other tectonically active regions. Nevertheless, a portion of intraplate earthquake activity in populated areas, including the NYCSZ, seems to be anthropogenic or suspected of being so. This earthquake activity is added onto the natural earthquake activity and is expected to be particularly shallow and close to people and thus more likely to be damaging (e.g., McGarr et a., 2002). It is also expected to increase as engineering operations increase in number and size. The largest instrumentally recorded earthquake in the NYCSZ, the 1994 M4.7 earthquake near Reading PA, was triggered by a quarry (Seeber et al., 1998). Available data worldwide show no evidence that the magnitude range of anthropogenic and of natural shallow intraplate earthquakes differ (e.g., Table 1). Liability issues are often of great concern and tend to interfere with the study of anthropogenic earthquake activity and its implications for hazard.

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Glossary of Key Terms

Anthropogenic: A byproduct of human activities. In the case of earthquakes, it refers to mechanical perturbations of the Earth's crust that could trigger fault rupture. In order to be significant, anthropogenic perturbations have to be larger than stress excursions associated with tides. Observations in tectonically active areas with intense earthquake activity show that static Coulomb stress perturbation (i.e., permanent changes in the elastic field) caused by large earthquakes start or enhance earthquake activity on some faults and decrease or shut down earthquake activity on others. These changes have been observed for calculated stress changes as small as 0.01MPa. A water-level change of only 1 meter generates a similar stress changes. Many types of engineering activities could thus generate significant stress perturbations. The following are often reported to trigger earthquakes: reservoir impounding; injection of fluids in deep wells; extensive mining of fluids, typically oil or water; deep underground cavities in mines; removal of surface load in quarries.

b-value: Slope of the line in log-linear plots describing the distribution of earthquakes over the magnitude range: $\log N = a - bm$. N is the cumulative number of earthquakes, i.e., with magnitudes $\geq m$, and a and b are constants, referred to as the "a-value" and "b-value". These constants are critical for hazard evaluations. The a-value measures the level of earthquake activity; the b-value establishes the expected rate of large damaging earthquakes in terms of the observed rate of much more numerous small earthquakes. Typically, there are about 10 earthquakes of magnitude m for each earthquake magnitude m+1, i.e., $b \approx 1$.

Coulomb Stress: A scalar assessment of the stress condition on a particular fault according to the Coulomb criterion for failure $\tau \geq \mu(\sigma_n - p)$. The shear and normal stress on the fault are τ and σ_n, respectively, p is the pore-fluid pressure, and μ is the coefficient of friction. The fault does not fail as long as the

Coulomb stress $\tau-\mu(\sigma_n-p)\leq 0$. A fault can be brought closer to failure by an increase in shear stress, or by a decrease in normal stress, or by an increase in pore pressure. Earthquakes may be triggered by perturbations of any of these parameters.

Craton: Geologically stable continental regions (SCR) that have not experienced extensional deformation since the last major pervasive compressional deformation event. This compression is typically associated with continental collision and growth by accretion. The compressional deformation is old, typically Precambrian, so that erosion has denuded the continent to near sea level and has exhumed rocks originally at mid-to-deep crustal levels. Historic data worldwide suggest that maximum earthquake size in cratons is in the magnitude 6-7 range. Instrumental data indicates that earthquake activity in cratons is concentrated in the upper few km of the crust.

Induced: An earthquake that results from a change of the Coulomb stress that is a substantial portion of the level of stress at failure. The upper SCR crust is generally close to failure, thus most significant earthquakes derived from anthropogenic perturbations are probably triggered. Small earthquakes may be induced locally near the sources of the perturbations, where the stress change may be large.

Intraplate: Within a plate, i.e., not a plate boundary. Intraplate areas include SCRs, as well as stable zones of oceanic crust as well as continental zones with significant diffused deformation, such as central Asia or western North America east of the San Andreas plate boundary.

Paleorift: Tectonically stable continental regions (SCR) that have experienced extensional deformation. Rifting precedes the current SCR regime, but postdates the last major pervasive compressional deformation event. Paleorifts are known or inferred to harbor large faults that accommodated the extension and rifting. The brittle upper crustal portion of the faults are typically preserved because little or no denudation has effected the continent after the rifting. All known SCR earthquakes in the magnitude ≥ 7 range are in these regions. Most of the

sources of deep crustal earthquake activity are also in these regions.

Stable continental regions (SCR) An intraplate continental area that exhibits little or no evidence of accumulated geologic deformation. SCRs incude cratons and paleorifts.

Stress Drop: The release in stress associated with slip on a fault. An earthquake transforms elastic strain energy into seismic energy, fracture energy, and heat. As a result, the stress generally decreases on the fault rupture and on parts of the surrounding rock. Stress drop refers to the average change of stress on the fault rupture.

Triggered: An earthquake that results from a perturbation of the Coulomb stress which is small to the pre-existing level of stress or to the drop in stress caused by the fault rupture associated with the earthquake. Earthquakes can be triggered by natural causes, such as other earthquakes, or by anthropogenic perturbations. Earthquakes can only be triggered on faults that are already close to failure. Stress changes as small as 0.01MPa (0.1 bars) are known to trigger earthquakes. The upper SCR crust appears to be generally close to failure, thus most significant earthquakes derived from anthropogenic perturbations are probably triggered.

TABLE 1

Mb≥6.0 or Ms≥6.0 or Surface Rupture in Stable Continental Regions; PDE: 1960-1990

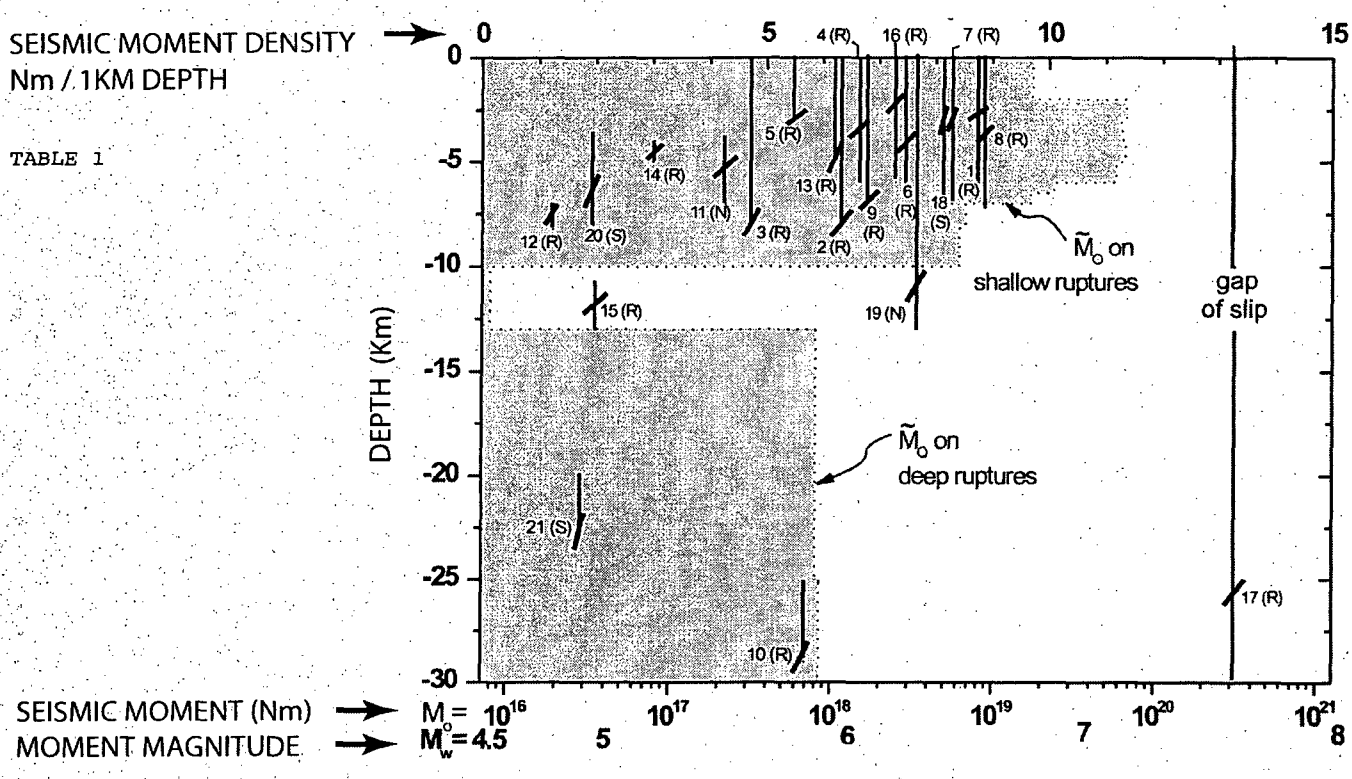
N	DATE	LAT.	LON.	Mb	Ms	RAKE degr.	DEPTH km			
Australia										
1.	1968 10 14	-31.518	116.971	6.0	6.8	62	(0-6)	CRA	SR	NT
2.	1970 03 10	-31.01	116.54	5.7		90?	(0-?)	CRA	SR	ET?
3.	1970 03 24	-21.981	126.682	6.2	5.9	80±5	8±3	CRA		NT
4.	1975 10 03	-22.126	126.721	6.0			C	CRA		ET?
5.	1979 06 02	-30.812	117.179	6.0	6.1	98±10	3±1	CRA	SR	ET?
6.	1986 03 30	-26.194	132.767	5.8	5.8	80±25	(0-3)	CRA	SR	?
7a.	1988 01 22	-19.798	133.910	6.1	6.3	90±10	6.5 4.5 2.0	CRA	SR	?
7b.	1988 01 22	-19.847	133.803	6.1	6.4	120±10	3.5 3.0 3.0	CRA	SR	ET
7c.	1988 01 22	-19.829	133.882	6.5	6.7	80±10	4.5 4.5 4.0	CRA	SR	ET
India										
8.	1967 12 10	+17.700	073.900	6.0		0±35	4.5	CRA	SR	T
9.	1993 09 29	+18.066	076.450	6.3	6.2	100	2.6 (0-6)	CRA	SR	T?
N America										
10.	1988 11 25	+48.050	288.900	5.9	6.0	60	(25-30)PR		NT	
11.	1989 12 25	+60.080	286.555	6.2	6.3	90	(0-3)	CRA	SR	NT
Africa										
12.	1982 12 22							PR?	SR	NT

1. 1968 Meckering	Gordon and Lewis, 1980; Vogfjord and Lagston, 1987
2. 1970 Calingiri	Gordon and Lewis, 1980
3. 1970 Canning Basin	Fredrich et al., 1988
4. 1975 Canning Basin?	PDE
5. 1979 Cadoux	Lewis et al., 1981; Fredrich et al., 1988
6. 1986 Marryatt Creek	Machette et al., 1993; Fredrich et al., 1988
7. 1988 Tennant Creek	Choy and Bowman, 1990; Crone et al., 1992
8. 1967 Koyna	Sahasrabudhe et al., 1969; Langston, 1981
9. 1993 Killari	Seeber et al., 1996; Baumbach et al., 1994
10. 1988 Saguenay	North et al., 1989
11. 1989 Ungava	Adams et al., 1991
12 1982 Guinea	Langer et al., 1987

CRA:	craton
PR:	paleorift zone
DEPTH:	centroid and (depth range) for rupture or subevents
C:	crustal depth poorly resolved
SR:	surface rupture

ET:	triggered by a previous earthquake
T:	anthropogenic triggering
NT:	no known possible cause of anthropogenic triggering
T?	possibility of triggering is being debated
?	triggering is not being investigated, but possible trigger is near epicenter.

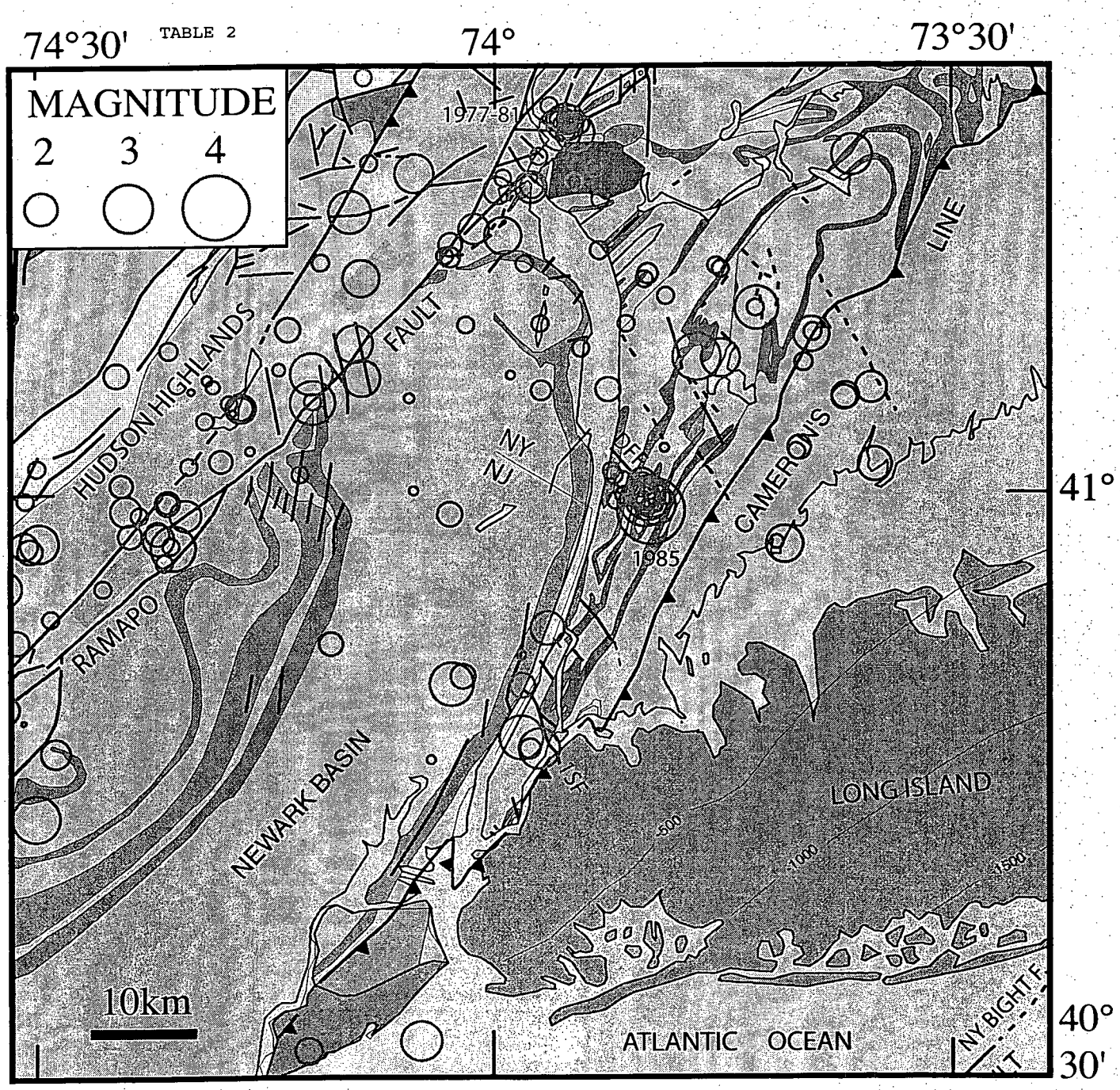
Note: Events 1, 2, 5 and events 3, 4 form spatial clusters similar to events 7a,b,c. They may be dependent sequences.



No.	SCR-earthquakes	Date dd/mm/yyyy	Rupture extension [km]	Focal depth [km]	Relative depth [%]	Crustal thickness [km]	Fault type	Seismic moment M_0 [Nm]	Dip angle [°]	References
Australia										
1	Meckering, AU	14/10/1968	0–6	3.0	19	32.5	R	8.20×10^{18}	29	Vogfjord and Langston (1987), Fredrich et al. (1988)
2	Lake McKay, AU	24/03/1970	0–8	8.0	25	32.5	R	1.17×10^{18}	45	Fredrich et al. (1988)
3	Simpson Desert, AU	28/08/1972	0–8	8.0	25	32.5	R	3.16×10^{17}	60	Fredrich et al. (1988)
4	Cadoux, AU	02/06/1979	0–6	4.0	19	32.5	R	1.49×10^{18}	34	Fredrich et al. (1988)
5	Marryat Creek, AU	30/03/1986	0–3	3.0	7	43.0	R	5.80×10^{17}	35	Fredrich et al. (1988)
6	Tennant 1, AU	22/01/1988	0–6	4.5	14	43.0	R	2.90×10^{18}	35	Choy and Bowman (1990)
7	Tennant 2, AU	22/01/1988	0–7	3.0	16	43.0	R	5.20×10^{18}	70	Choy and Bowman (1990)
8	Tennant 3, AU	22/01/1988	0–7	4.5	16	43.0	R	8.30×10^{16}	45	Choy and Bowman (1990)
North America										
9	Baffin Bay, Canada	04/09/1963	0–7	7.0	19	37.5	R	1.70×10^{18}	41	Hasegawa and Adams (1990)
10	Saguenay, Canada	25/11/1988	25–30	29	77	37.5	R	6.90×10^{17}	67	North et al. (1989)
11	Miramichi, Canada	09/01/1982	3.5–7		19	37.5	N	2.20×10^{17}	50	Wetmiller et al. (1984)
12	Goodnow, USA	07/10/1983	7–8	7.5	22	37.5	R	1.90×10^{16}	60	Nabelek and Suarez (1989)
13	Ungava, Canada	25/12/1989	0–5	5.0	13	37.5	R	1.10×10^{18}	70	Adams et al. (1991)
14	Pymatuning, USA	25/09/1998	4–5	4.5	13	37.5	R	1.00×10^{17}	65	Seeber (pers. com.)
15	Au Sable Forks, USA	20/04/2002	10–13	11.5	35	37.5	R	3.50×10^{16}	45	Seeber et al. (2002)
Asia										
16	Killari	29/09/1993	0–6	2.6	16	37.5	R	1.70×10^{18}	46	Seeber et al. (1996)
17	Bhuj	26/01/2001	0–10,13–30	26.0	80	37.5	R	3.16×10^{19}	41	Singh et al. (2004), Bodin and Horton (2004)
Africa										
18	Ceres, RSA	29/09/1969	0–6.5	4.0	16	40.0	S	5.01×10^{18}	87	Green and Bloch (1971)
19	West Guinea	22/12/1983	0–13	11.0	32	35.0	N	3.40×10^{18}	60	Langer et al. (1987)
Europe										
20	Schwabian Jura, D	03/09/1978	3–7.5	6.5	27	30.0	S	3.40×10^{16}	85	Haessler et al. (1980), Scherbaum et al. (1983)
21	North Wales, UK	19/07/1984	20–23	23.0	67	30.0	S	2.24×10^{17}	79	Ansell et al. (1986)

Figure 1. Depth range of ruptures (vertical lines), focal depth, and seismic moment (& moment magnitude), of 21 intraplate continental earthquakes, 4.5<Mw<8. Oblique lines indicate focal depth (rupture initiation) and faut dip. R=reverse, S=strike-slip, N=normal. The line bounding the shaded area shows the crustal depth distributionof the seismic moment density (in %/ per 1km depth). From Klose and Seeber, 2007.

TABLE 2



Cretaceous and younger; Coastal Plain deposits

Triassic-Jurassic sediments and basic intrusives

Devonian sediment (Hamilton Group)

Ordovician intrusives

Ordovician shale and siltstone

Ordovician-Cambrian Manhattan schist with serpentine bodies

Ordovician-Cambrian Limestone /marble

Precambrian gneiss and granite

Faults
barbs on hangingwall of thrusts

-500 Bedrock depth contours in feet

Figure 2: Bedrock map of the New York City area and epicenters (1975-2002). The 1985 earthquake occurred on the Dobbs Ferry fault (DFF), which is one of a family of NW-striking faults that include the 125th Street fault in Manhattan (1SF)

LEONARDO SEEBER

Professional Preparation:
Nuclear Engineering, Columbia University, BS 1965
Graduate Student in Seismology, Lamont-Doherty Geological Observatory, 1965-1969

Appointments:
Doherty Senior Research Scientist, Lamont-Doherty Earth Observatory, 2003-present
Doherty Research Scientist, Lamont-Doherty Earth Observatory, 1998-2003
Associate Research Scientist, Lamont-Doherty Earth Observatory, 1983-1998
Research Associate, Lamont-Doherty Geological Observatory, 1979-1983
Research Scientist, Lamont-Doherty Geological Observatory, 1972-1979
Self-Employed Grape Farmer, Central Italy, 1969-1973

Related Publications:
2004 Seeber, L., J.G. Armbruster, and W.-Y. Kim, A fluid-injection triggered earthquake sequence in Ashtabula OH: Implications for seismogenesis in stable continental regions, Bull. Seism. Soc. Am. 94, 76-87.
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Significant Recent Publications:

2007, Seeber, L., C. Mueller, T. Fujiwara, K. Arai, W. Soh, Y. S. Djajadihardja, M.-H. Cormier, Accretion, mass wasting, and partitioned strain over the 26 December 2004 M9.2 rupture offshore Aceh, northern Sumatra, accepted Earth & Planetary Science Letters.

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Seeber, L. and J.G. Armbruster, The San Andreas fault system through the eastern Transverse Ranges as illuminated by earthquakes. Jour. Geoph. Res, 100, p. 8285-8310, 1995.

Synergistic Activities:

•2001-2005: Steering Committee of the Center for Hazard and Risk Research of the Columbia University Earth Institute; Application to the reduction of earthquake risk in Istanbul, Turkey.

•1996-1999 Co-PI in NSF-funded project targeting High School curricula: Earth-View Explorer: An Interactive Educational Tool to Explore Topics in Earth Science;

•1995-1998 Member of the International Advisory Committee to the Government of India in matters of earthquake monitoring and Hazard.

•1992-2002 Member Board of Directors Southern California Earthquake Center;

•1998-1999 Conceptualization and Design of exhibits for the Hall of the Earth in the American Museum of Natural History

•1990-2000 Development of a hypocenter, focal mechanism, and slip plane data base for southern California from phase data of the southern California Network (*Anonymous FTP*: scec.gps.caltech.edu; *Directory:* /pub/focal; *File:* focal1.nano).

•2006 Member of Reconstruction Assistance Team of the US National Academies, following the destructive 2005 earthquake in northern Pakistan

Current Collaborators:
Omer Emre (MTA, Turkey); Aral Okay (ITU Turkey); Sue Hough (USGS, Pasadena, CA); Alessandro Amato (INGV Rome); Ignazio Guerra (Universita della Calabria). Michael Steckler (L-DEO); Milene Cormier (Univ. of Missouri); Cecilia McHugh (Queens College); Colin Stark (L-DEO); Thorsten Nagel, (Univ. of Bonn); Andrea Taramelli (Univ. of Perugia); Luc Lavier, (Univ. of Texas).

Advisees: Students (* graduated) **and Postdocs** (**):
Patience Cowie (University of Edinburg, GB)*; Nancye Dawers (U. of Texas, Austin, TX) *, Ken Hudnut* (USGS, Pasadena); Stephen Jaumé* (University of Charleston, SC); Craig Nicholson* (UCSB, Santa Barbara, CA); Marc Petersen* (USGS, Golden, CO); Jan Vermilie** (Whittier College, Whittier, CA), Patrick Williams * (Univerity of Oregon). Teresa Lawler, Christian Close**, Alessandro Sorichetta** (Universita di Camerino and Queens College); Meg Reitz (Columbia Univ.)