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June 25, 18

Chairman Kevin J. McIntyre
Federal Energy Regulatory Commission
888 First Street, N.E.
Washington DC 20426

Subject: Safety Study of Algonquin Natural Gas Pipelines, Westchester County, New York

Dear Chairman McIntyre:

I have received a copy of the letter from the State of New York agencies regarding the risk assessment of the AIM pipelines in the vicinity of the Indian Point Nuclear power plants. The letter stated, "FERC must re-evaluate whether the NRC and Entergy analyses relied on by FERC during the review of the AIM project were sufficient. The NRC and Entergy analyses concluded that the IPEC reactors could safely shut down if there were a pipeline incident, but it may not have fully considered all necessary and appropriate factors."

The letter affirms the points I have raised in filings to the AIM docket and in correspondence to FERC, PHMSA and the NRC since 2014. All of these documents are available from these agencies or upon request. As a nuclear expert, I have consistently stated that the NRC and Entergy analysis was fundamentally flawed and that FERC should require an independent risk assessment. This was never done and the pipeline was constructed and completed and put into service in January 2017. I urge you to immediately require the production of a risk assessment conducted as required by 49 CFR 192.917 (Enclosure 1) of the AIM pipeline because of the unacceptable risk to the 20 million people within the 50-mile radius of Indian Point.

The recent letter from the New York State agencies to FERC identified some of the shortcomings of the design and construction of the AIM pipeline but falls short of identifying all of the requirements of 49 CFR 192, specifically compliance with ASME B31.8(s), incorporated by reference, and the requirements for public awareness and first responder training and awareness also required by 49 CFR 192.

A specific example of one shortcoming is the rupture of the pipeline and the impact on the control room and operators. There is no capability within the control room to detect the presence of a toxic and explosive gas presence. A second shortcoming in the State summary is the recommendation to test isolation valves to assure a 3-minute closure time. This is meaningless unless operator response time is included. Additionally, the assumed Potential Impact Radius[1] (PIR) only considers radiant heat flux and fails to consider vapor clouds and atmospheric conditions such as wind speed and direction. The EPA ALOHA program even predicts blast and fire radius extending out to one mile under certain atmospheric conditions. Moreover, using the equation provided by the NRC in Regulatory Guide 1.91, three professional engineers have calculated a PIR exceeding 4000 feet. These are a few shortcomings and in the complete report there may be more. While the ALOHA program is not recommended for analyzing pipe breaks on long sections of pipes, it is very useful predicting the impact of vapor clouds and resulting ignitions considering local atmospheric conditions.

I will be requesting via separate letter, a copy of the entire report provided to FERC by the State of New York. I have been previously been approved by FERC to receive such CEII information for this docket.

My specific area of expertise is nuclear power safety and regulations and I provided expert consulting services for the New York Attorney General (Cuomo) related to the proposed relicensing of Indian Point. It was during this timeframe I identified the potential risk presented by the existing pipelines. I identified this problem to the Attorney General's office; however, it was ignored.

A risk assessment for most proposed high hazard projects is required by federal regulations. For example proposed nuclear power plant projects are required to produce a risk assessment under the requirements of 10 CFR 50.34 (Enclosure 2). This is a monumental effort and requires hundreds of person-years of effort. The outcome of this risk assessment is referred to as the Final Safety Analysis Report (FSAR), is publicly available and addresses each requirement of 10 CFR 50.

The requirements of 10 CFR 50 are very similar to the requirements of 49 CFR 192, however FERC and PHMSA appear to totally disregard any attempt to comply with the requirements of 49 CFR 192.917 for a detailed risk assessment. I have attached these requirements as Enclosures 1 and 2.

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[1] A Model For Sizing High Consequence Areas Associated With Natural Gas Pipelines, Mark J. Stephens, C-FER Technologies, C-FER Report 99068. October, 2000.

The NRC enforces its risk regulations at a cost of hundreds of millions of dollars, whereas PHMSA and FERC just totally ignore these requirements risk assessment requirements and endangering the lives of millions of residents within more than a mile of the pipelines and the State of New York tacitly approves this approach.

The requirements are very clear within 49 U.S.C. 60109 (High-density population areas and Environmentally sensitive areas) and the NEW York PSC should have recognized this failure and shortcoming with the initiation of the proposed AIM project. This regulation clearly states:

(C) Transmittal of programs to state authorities.—
The Secretary shall provide a copy of each risk analysis and integrity management program reviewed by the Secretary under this paragraph to any appropriate State authority with which the Secretary has entered into an agreement under section 60106.

Responses to my FOIA to PHMSA and the State of New York requests confirm this was never accomplished.

FERC's approval of this project was predicated on numerous statements by PHMSA and Spectra that the project would/will be in compliance with 49 CFR 192 without any identified exceptions.

49 CFR 192.917[2] is very clear that this is a firm requirement. Numerous FOIA requests and appeals to PHMSA, however, have determined that this requirement has not been fulfilled. Therefore, the final approval by FERC was based on apparently inaccurate and false statements by Spectra and PHMSA contained within the initial application, the DEIS and FEIS.

I am also enclosing a copy of my testimony and letter to FERC of September 27, 2014 (Enclosure 3) and my letter to the FERC Chairman dated 12/17/2015 (Enclosure 4) outlining the shortcomings of the FERC process for the AIM review that were ignored in FERC's approval of the AIM project. The total lack of disregarding these comments further contributes to the risks with these installed lines.

While I take no formal position, many of my colleagues believe FERC must revoke its permit and stop the flow of gas in the vicinity of Indian Point until the

[2] https://www.law.cornell.edu/cfr/text/49/192.917

required risk assessment has been completed and all permits have been reconsidered and issued based on the corrected information and process.

With more than 45 years of dealing with federal regulatory agencies I have never observer such a blatant and apparent deliberate disregard of the United States Code (U.S.C) and the Code of Federal Regulations (CFR). It is my professional opinion that FERC and PHMSA have the statutory responsibility to require and assure compliance with federal laws and regulations. It is very clear from my numerous FOIA requests to PHMSA and previous letters to the FERC Chairpersons, that these requirements have not been met therefore FERC must immediately impose the requirements of 49 CFR 192.917 requirements and take immediate protective actions to assure public safety by imposing restriction on pipeline operation.

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Enclosure 1
PHMSA Requirements for Risk Assessment

§ 192.917 How does an operator identify potential threats to pipeline integrity and use the threat identification in its integrity program?

(a)*Threat identification.* An operator must identify and evaluate all potential threats to each covered pipeline segment. Potential threats that an operator must consider include, but are not limited to, the threats listed in ASME/ANSI B31.8S (incorporated by reference, *see*§ 192.7), section 2, which are grouped under the following four categories:

(1) Time dependent threats such as internal corrosion, external corrosion, and stress corrosion cracking;

(2) Static or resident threats, such as fabrication or construction defects;

(3) Time independent threats such as third party damage and outside force damage; and

(4) Human error.

(b)*Data gathering and integration.* To identify and evaluate the potential threats to a covered pipeline segment, an operator must gather and integrate existing data and information on the entire pipeline that could be relevant to the covered segment. In performing this data gathering and integration, an operator must follow the requirements in ASME/ANSI B31.8S, section 4. At a minimum, an operator must gather and evaluate the set of data specified in Appendix A to ASME/ANSI B31.8S, and consider both on the covered segment and similar non-covered segments, past incident history, corrosion control records, continuing surveillance records, patrolling records, maintenance history, internal inspection records and all other conditions specific to each pipeline.

(c)*Risk assessment.* An operator must conduct a risk assessment that follows ASME/ANSI B31.8S, section 5, and considers the identified threats for each covered segment. An operator must use the risk assessment to prioritize the covered segments for the baseline and continual reassessments (§§ 192.919, 192.921, 192.937), and to determine what additional preventive and mitigative measures are needed (§ 192.935) for the covered segment.

(d)*Plastic transmission pipeline.* An operator of a plastic transmission pipeline must assess the threats to each covered segment using the information in sections 4 and 5 of ASME B31.8S, and consider any threats unique to the integrity of plastic pipe.

(e)*Actions to address particular threats.* If an operator identifies any of the following threats, the operator must take the following actions to address the threat.

(1)*Third party damage.* An operator must utilize the data integration required in paragraph (b) of this section and ASME/ANSI B31.8S, Appendix A7 to determine the susceptibility of each covered segment to the threat of third party damage. If an operator identifies the threat of third party damage, the operator must implement comprehensive additional preventive measures in accordance with § 192.935 and monitor the effectiveness of the preventive measures. If, in conducting a baseline assessment under § 192.921, or a reassessment under § 192.937, an operator uses an internal inspection tool or external corrosion direct assessment, the operator must integrate data from these assessments with data related to any encroachment or foreign line crossing on the covered segment, to define where potential indications of third party damage may exist in the covered segment.

An operator must also have procedures in its integrity management program addressing actions it will take to respond to findings from this data integration.

(2)*Cyclic fatigue.* An operator must evaluate whether cyclic fatigue or other loading condition (including ground movement, suspension bridge condition) could lead to a failure of a deformation, including a dent or gouge, or other defect in the covered segment. Anevaluation must assume the presence of threats in the covered segment that could be exacerbated by cyclic fatigue.

An operator must use the results from the evaluation together with the criteria used to evaluate the significance of this threat to the covered segment to prioritize the integrity baseline assessment or reassessment.

(3)*Manufacturing and construction defects.* If an operator identifies the threat of manufacturing and construction defects (including seam defects) in the covered segment, an operator must analyze the covered segment to determine the risk of failure from these defects. The analysis must consider the results of prior assessments on the covered segment. An operator may consider manufacturing and construction related defects to be stable defects if the operating pressure on the covered segment has not increased over the maximum operating pressure experienced during the five years preceding identification of the high consequence area. If any of the following changes occur in the covered segment, an operator must prioritize the covered segment as a high risk segment for the baselineassessment or a subsequent reassessment.

(i) Operating pressure increases above the maximum operating pressure experienced during the preceding five years;

(ii) MAOP increases; or

(iii) The stresses leading to cyclic fatigue increase.

(4)*ERW pipe.* If a covered pipeline segment contains low frequency electric resistance welded pipe (ERW), lap welded pipe or other pipe that satisfies the conditions specified in ASME/ANSI B31.8S, Appendices A4.3 and A4.4, and any covered or noncovered segment in the pipeline system with such pipe has experienced seam failure, or operating pressure on the covered segment has increased over the maximum operating pressure experienced during the preceding five years, an operator must select an assessment technology or technologies with a proven application capable of assessing seam integrity and seam corrosion anomalies. The operator must prioritize the covered segment as a high risk segment for the baseline assessment or a subsequent reassessment.

(5)*Corrosion.* If an operator identifies corrosion on a covered pipeline segment that could adversely affect the integrity of the line (conditions specified in § 192.933), the operator must evaluate and remediate, as necessary, all pipeline segments (both covered and non-covered) with similar material coating and environmental characteristics. An operator must establish a schedule for evaluating and remediating, as necessary, the similar segments that is consistent with the operator's established operating and maintenance procedures under part 192 for testing and repair.

Enclosure 2
NRC Requirements for a Risk Assessment

§ 50.34 Contents of applications; technical information.

(a) *Preliminary safety analysis report*. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[5] to be included shall consist of the following:
(1) Stationary power reactor applicants for a construction permit who apply on or after January 10, 1997, shall comply with paragraph (a)(1)(ii) of this section. All other applicants for a construction permit shall comply with paragraph (a)(1)(i) of this section.
(i) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in part 100 of this chapter. The assessment must contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.
(ii) A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:
(A) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;
(B) The extent to which generally accepted engineering standards are applied to the design of the reactor;
(C) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;
(D) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:
(*1*) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[7] total effective dose equivalent (TEDE).
(*2*) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE);
(E) With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by paragraph (a)(1)(i) of this section, in support of the application for a construction permit.
(2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.
(3) The preliminary design of the facility including:
(i) The principal design criteria for the facility.[8] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units;
(ii) The design bases and the relation of the design bases to the principal design criteria;
(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.
(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling

performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.
(5) An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design: *Provided, however*, That this requirement is not applicable to an application for a construction permit filed prior to January 16, 1969.
(6) A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.
(7) A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for quality assurance programs for nuclear power plants and fuel reprocessing plants. The description of the quality assurance program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied.
(8) An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.
(9) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.
(10) A discussion of the applicant's preliminary plans for coping with emergencies. Appendix E sets forth items which shall be included in these plans.
(11) On or after February 5, 1979, applicants who apply for construction permits for nuclear powerplants to be built on multiunit sites shall identify potential hazards to the structures, systems and components important to safety of operating nuclear facilities from construction activities. A discussion shall also be included of any managerial and administrative controls that will be used during construction to assure the safety of the operating unit.
(12) On or after January 10, 1997, stationary power reactor applicants who apply for a construction permit, as partial conformance to General Design Criterion 2 of appendix A to this part, shall comply with the earthquake engineering criteria in appendix S to this part.
(13) On or after July 13, 2009, stationary power reactor applicants who apply for a construction permit shall submit the information required by 10 CFR 50.150(b) as a part of their preliminary safety analysis report.
(b) *Final safety analysis report*. Each application for an operating license shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:
(1) All current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issuance of the construction permit, relating to site evaluation factors identified in part 100 of this chapter.
(2) A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, 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.
(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.
(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.
(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter.
(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974.
(5) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved.
(6) The following information concerning facility operation:
(i) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.
(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power

plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied.
(iii) Plans for preoperational testing and initial operations.
(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.
(v) Plans for coping with emergencies, which shall include the items specified in appendix E.
(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36.
(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites.
(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.
(8) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of part 55 of this chapter.
(9) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in § 50.61 (b)(1) and (b)(2).
(10) On or after January 10, 1997, stationary power reactor applicants who apply for an operating license, as partial conformance to General Design Criterion 2 of appendix A to this part, shall comply with the earthquake engineering criteria of appendix S to this part. However, for those operating license applicants and holders whose construction permit was issued before January 10, 1997, the earthquake engineering criteria in Section VI of appendix A to part 100 of this chapter continues to apply.
(11) On or after January 10, 1997, stationary power reactor applicants who apply for an operating license, shall provide a description and safety assessment of the site and of the facility as in § 50.34(a)(1)(ii). However, for either an operating license applicant or holder whose construction permit was issued before January 10, 1997, the reactor site criteria in part 100 of this chapter and the seismic and geologic siting criteria in appendix A to part 100 of this chapter continues to apply.
(12) On or after July 13, 2009, stationary power reactor applicants who apply for an operating license which is subject to 10 CFR 50.150(a) shall submit the information required by 10 CFR 50.150(b) as a part of their final safety analysis report.
(c) *Physical Security Plan*. (1) Each applicant for an operating license for a production or utilization facility that will be subject to §§ 73.50 and 73.60 of this chapter must include a physical security plan.
(2) Each applicant for an operating license for a utilization facility that will be subject to the requirements of § 73.55 of this chapter must include a physical security plan, a training and qualification plan in accordance with the criteria set forth in appendix B to part 73 of this chapter, and a cyber security plan in accordance with the criteria set forth in § 73.54 of this chapter.
(3) The physical security plan must describe how the applicant will meet the requirements of part 73 of this chapter (and part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a) of this chapter, at the proposed facility). Security plans must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR parts 11 and 73, if applicable.
(d) *Safeguards contingency plan*. (1) Each application for a license to operate a production or utilization facility that will be subject to §§ 73.50 and 73.60 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in section I of appendix C to part 73 of this chapter. The "implementation procedures" required per section I of appendix C to part 73 of this chapter do not have to be submitted to the Commission for approval.
(2) Each application for a license to operate a utilization facility that will be subject to § 73.55 of this chapter must include a licensee safeguards contingency plan in accordance with the criteria set forth in section II of appendix C to part 73 of this chapter. The "implementing procedures" required in section II of appendix C to part 73 of this chapter do not have to be submitted to the Commission for approval.
(e) *Protection against unauthorized disclosure*. Each applicant for an operating license for a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, a training and qualification plan, or a cyber security plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter.
(f) *Additional TMI-related requirements*. In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982, shall meet the requirements in paragraphs (f)(1) through (3) of this section. This regulation applies to the pending applications by Duke Power Company (Perkins Nuclear Station, Units 1, 2, and 3), Houston Lighting & Power Company (Allens Creek Nuclear Generating Station, Unit 1), Portland General Electric Company (Pebble Springs Nuclear Plant, Units 1 and 2), Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Puget Sound Power & Light Company (Skagit/Hanford Nuclear Power Project, Units 1 and 2), and Offshore Power Systems (License to Manufacture Floating Nuclear Plants). The number of units that will be specified in the manufacturing license above, if issued, will be that number whose start of manufacture, as defined in the license application, can practically begin within a 10-year period commencing on the date of issuance of the manufacturing license, but in no event will that number be in excess of ten. The manufacturing license will require the plant design to be updated no later than 5 years after its approval. Paragraphs (f)(1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this regulation. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and criterion 50 of appendix A to 10 CFR part 50, is sufficient for

issuance of that manufacturing license which may be considered in the manufacturing license proceeding. In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of these studies are factored into the final design of the facility. For licensees identified in the introduction to paragraph (f) of this section, all studies must be completed no later than 2 years following the issuance of the construction permit or manufacturing license.[10] For all other applicants, the studies must be submitted as part of the final safety analysis report.[10]

(i) Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)

(ii) Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only) (II.E.1.1):

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(B) A design review of AFWS.

(C) An evaluation of AFWS flow design bases and criteria.

(iii) Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

(iv) Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)

(v) Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.13)

(vi) Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)

(vii) Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

(viii) Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.24)

(x) Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

(xi) Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)

(xii) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.

(i) Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only) (I.A.4.2.)

(ii) Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures,

reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)
(iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)
(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)
(v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)
(vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)
(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term1A[11]radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)
(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term1A[11] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)
(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8)
(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.
(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.
(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.
(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.
(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)
(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)
(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)
(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)
(xiv) Provide containment isolation systems that: (II.E.4.2)
(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,
(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,
(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,
(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,
(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.
(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)
(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)
(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)
(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)
(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)
(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)
(xxii) Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (II.K.2.9)
(xxiii) Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (II.K.2.10)
(xxiv) Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)
(xxv) Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2).
(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term1A[11]radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)
(xxvii) Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)
(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems. (III.D.3.4)
(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.
(i) Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)
(ii) Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)
(iii) Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)
(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)
(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)
(A)(*1*) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.
(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852-2738.
(B)(*1*) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, subarticle CC-

3720, Service Load Category, (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.
(vi) For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)
(vii) Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)
(g) *Combustible gas control.* All applicants for a reactor construction permit or operating license whose application is submitted after October 16, 2003, shall include the analyses, and the descriptions of the equipment and systems required by § 50.44 as a part of their application.
(h) *Conformance with the Standard Review Plan (SRP).* (1)(i) Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.
(ii) Applications for light-watercooled nuclear power plant construction permits docketed after May 17, 1982, shall include an evaluation of the facility against the SRP in effect on May 17, 1982, or the SRP revision in effect six months before the docket date of the application, whichever is later.
(2) The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.
(3) The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.
(i) A description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to explosions or fire as required by § 50.54(hh)(2) of this chapter.
Editorial Note: For Federal Register citations affecting § 50.34, see the List of CFR Sections Affected.
[5] The applicant may provide information required by this paragraph in the form of a discussion, with specific references, of similarities to and differences from, facilities of similar design for which applications have previously been filed with the Commission.
[6] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.
[7] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.
[8] General design criteria for chemical processing facilities are being developed.
[9] [Reserved].
[10]Alphanumeric designations correspond to the related action plan items in NUREG 0718 and NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." They are provided herein for information only.
[11] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.
[33 FR 18612, Dec. 17, 1968; 72 FR 49491, Aug. 28, 2007; 73 FR 63571, Oct. 24, 2008; 74 FR 13969, Mar. 27, 2009; 74 FR 28146, Jun. 12, 2009; 80 FR 74980, Dec. 1, 2015; 81 FR 86909, Dec. 2, 2016]
Page Last Reviewed/Updated Tuesday, August 29, 2017

Enclosure 3
9/27/2014 Letter to
FERC

Paul M. Blanch

Energy Consultant

June 25, 2018

Kimberly D. Bose, Secretary
Federal Energy Regulatory Commission
888 First Street NE, Room 1A
Washington, DC 20426

Subject: Algonquin Gas Transmission, LLC
Docket No. CP14-96-000
FERC/EIS-0254D

Dear Ms. Bose

I am submitting the following comments on behalf of myself on the above-proposed project. I am a registered Professional Engineer with more than 45 years of Nuclear and Federal regulatory requirements.

I have been a consultant to the Chief Nuclear Officers at Indian Point and also an expert witness for the Attorney General for the State of New York related to the relicensing efforts of Indian point.

In October 2010[3] I petitioned[4] the Nuclear Regulatory Commission (NRC) to evaluate the risks associated with the existing gas lines. The NRC in its response stated this analysis had been conducted however they would not share it with me due to national security concerns

I have conducted a detailed review of the Draft Environmental Impact Statement (DEIS) and the requirements as stated in 49 CFR 192 and also 30 CFR Part 380, Appendix A to Part 380 – "Minimum Filing Requirements for Environmental Reports Under the Natural Gas Act."

Based upon these Federal requirements I have the following comments related to the DEIS:

1. 30 CFR 380 (m), Reliability and Safety explicitly states:

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[3] http://pbadupws.nrc.gov/docs/ML1030/ML103020293.pdf
[4] http://www.huffingtonpost.com/huff-wires/20101025/us-indian-point-gas-line/

"Describe how the project facilities would be designed, constructed, operated, and maintained to minimize potential hazard to the public from the failure of project components as a result of accidents or natural catastrophes. (§ 380.12(m))."

This proposed line is located in the vicinity of residents, schools, churches and one of the largest nuclear plants in the USA. 49 CFR 192 discusses various design requirements for safety. I note that the new lines are not designed to the most stringent safety requirements of Class 4 lines. Contrary to these requirements I did not see any discussion within the DEIS or the application discussing what provisions would be incorporated to minimize the impact to the public and why these lines are not designed to the maximum safety standards specified by 49 CFR 192.111 and 49 CFR 192.5. These standards would require closer isolation valve spacing, and more robust pipes designed to withstand higher pressures. While not a specific requirement to design these lines as Class 4, it was never anticipated that gas transmission lines would be located near or on the property of a nuclear power facility.
There is no discussion in either the AIM proposed description or the DEIS as to automatic isolation valves which had been remover from the original gas lines. The only isolation valves are controlled from Houston, Texas and there is no assurance these will be operable due to an earthquake or other natural disaster.

2. The White Plains Journal News published the following Community View on September 15, 2014. Some of these may be new issues however none of these issues have been addressed in either the DEIS or the Spectra Application.

"View: Algonquin plan poses risks to Indian Point, residents

Paul Blanch 10 p.m. EDT September 14, 2014 Spectra plans to place a larger gas pipeline near Indian Point. The probability of a gas line failure is remote but is not zero. It is unconscionable and irresponsible to continue this project prior to a complete, independent risk analysis.

Nuclear power plants and natural gas transmission lines provide energy for homes and businesses. Due to the inherent hazards associated with these energy sources, the federal government "regulates" both. The proposed routing of the Algonquin natural gas pipeline near the Indian Point nuclear plant poses the risk that these hazards may team up to harm the community.
I speak as a professional engineer with more than 45 years of nuclear experience including formerly reporting directly to the Chief Nuclear Officer at Indian Point and an expert witness for the State of New York related to the relicensing of Indian Point.

There are three gas existing natural gas transmission lines traversing the Indian Point site within 600 feet of vital structures. There has not been any publicly available analysis demonstrating the risks of these lines. The Nuclear Regulatory Commission has refused to provide this information under the guise of national security, yet has maintained the "secret" analysis shows Indian Point is not at undue risk.

Failure of any of these lines could result in a total loss of cooling to the reactor cores and 40 years inventory of spent fuel. There are no provisions within the area to combat this event until valves are remotely closed from the pipeline company's facility in Houston, Texas. In the meantime, the energy released from a ruptured line in one hour would exceed the energy released from one of the atomic bombs dropped on Japan in 1945.

Some of the possible consequences of a gas line fire/explosion to Indian Point include loss of power to the entire site, secondary fires from liquid fuel storage tanks, reactor core damage and melting, asphyxiation

of site personnel, spent fuel radioactivity releases exceeding those of Fukushima, and social/economic damages exceeding $1 trillion.

Now Algonquin/Spectra wants to place yet another high-pressure 42-inch line also in the vicinity of Indian Point, doubling the existing capacity. According to the Federal Energy Regulatory Commission, "the proposed route would not pose any new hazard to the (Indian Point) facility." There is no way FERC could make this determination without a complete risk analysis. And FERC's Draft Environmental Impact Statement ignores damage prevention, emergency response and public awareness, which are federal Department of Transportation requirements.

Algonquin gas pipeline project sparks safety concerns

An independent study of a gas pipeline near a nuclear facility in another state concluded it represented an undue risk. The amount of gas flow and energy in that pipeline was less than 1/1000 of the Algonquin/Spectra project and the facility was located in an area with much lower population.

The probability of a gas line failure is remote but is not zero especially if terrorism is considered. This may possibly be one of the most attractive targets in the nation.

The event would be aggravated by the decision of Spectra to not include any automatic gas termination valves and no means to combat the fire/explosion prior to gas flow termination. The gas lines are not designed to the most stringent safety standards as discussed in DOT regulations. The only gas isolation valves are remotely controlled from Houston, Texas. It seems the community around Indian Point is protected against a gas pipeline rupture triggering a nuclear plant accident—unless a gas pipeline ruptures. That's unacceptable.

The State of New York and all of the impacted counties must demand an independent and transparent analysis be conducted by an independent engineering organization. The cost for this study should be borne by Spectra/Entergy.

It is unconscionable and irresponsible to continue this project prior to a complete, independent risk analysis. The potential consequences of this event are too devastating to the New York area and my home State of Connecticut not to design this new line to maximum safety standards and assess the risk.

The writer, a West Hartford, Conn., resident, is an engineer."

30 CFR Part 380 also requires:

(1) Describe measures proposed to protect the public from failure of the proposed facilities (including coordination with local agencies).

(3) Discuss design and operational measures to avoid or reduce risk.

(5) Describe measures used to exclude the public from hazardous areas. Discuss measures used to minimize problems arising from malfunctions and accidents (with estimates of probability of occurrence) and identify standard procedures for protecting services and public safety during maintenance and breakdowns.

Again, none of these requirements met or addressed.

3. Page ES-8 FERC DEIS FERC states:

"Algonquin identified that because of the distance of the proposed Project from the IPEC generating facilities and the avoidance and mitigation measures that it would implement, the proposed ***route would not pose any new safety hazards to the IPEC***

facility. *To ensure that the AIM Project would not present new safety hazards to the IPEC facility, we are recommending that Algonquin file the final conclusions regarding any potential safety-related conflicts with the IPEC based on the Hazards Analysis performed by Entergy."*

This is one of the most egregious statements within the DEIS and is an irresponsible and rash statement with no bases. The Nuclear Regulatory Commission (NRC) has reviewed similar analysis at nuclear facilities nuclear facilities with 1/1000 of the proposed gas flow and located more than one mile from the facility and determined that a 16-inch operating at 50-PSI. The study performed by Framatome determined gas line presented undue risk to the facility. Any analysis conducted with a foregone outcome as stated within the DEIS is completely unscientific and irresponsible. It should be FERC's responsibility to assure this analysis is conducted in an open, scientific, transparent independent manner with a peer review. This analysis cannot be conducted by any organization with a vested interest such as Spectra/Algonquin, Indian Point/Entergy or the Nuclear Regulatory Commission.

West Point Partners, LLC ("WPP") proposes to construct and operate the West Point Transmission Project ("the Project"), an approximately 80-mile-long high voltage electric transmission facility that will connect the existing National Grid Leeds Substation (Leeds Substation) in the Town of Athens, Greene County, NY, and the existing Consolidated Edison Company of New York, Inc. (Con Edison), Buchanan North Substation (Buchanan Substation) located adjacent to the Indian Point Energy Center in the Village of Buchanan, Town of Cortlandt, Westchester County, NY. For approximately 77 miles of its length, the Project will be buried under the bed of the Hudson River.

Both the American Society of Civil Engineers[5] and the National Association of Corrosion Engineers clearly state[6] that high voltage direct current (HVDC) lines will induce "stray currents" which will accelerate the corrosion of metallic piping systems. This HVDC line will directly intersect with both the new and 60 year old degrading existing gas transmission lines and piping systems and tanks at the Indian Point facility.

49 CFR Part 192, Appendix D to Part 192 - Criteria for Cathodic Protection and Determination of Measurements require this to be addressed and measures implemented to assure that there will be no impact or stray current corrosion induced by the HVDC lines in the proximity of the gas lines..

[5] http://ascelibrary.org/doi/abs/10.1061/9780784413142.093
[6] http://www.nace.org/cstm/Store/Product.aspx?id=b7a6056e-bb57-df11-a321-005056ac759b

4. 49 CFR 192.615[7] requires "each operator shall establish written procedures to minimize the hazard resulting from a gas pipeline emergency."

   There is no discussion within the DEIS as to how this problem will be addressed especially when remotely operated valves are controlled from Houston, Texas.

5. 49 CFR §192.616 Public awareness requires "each pipeline operator must develop and implement a written continuing public education program that follows the guidance provided in the American Petroleum Institute's (API) Recommended Practice (RP) 1162 (incorporated by reference, *see* §192.7)."

   There is no discussion within the DEIS of the application as to how this is being addressed. This public education process must include the potential consequences of impact to the Indian Point nuclear plants and how an accident would be minimized.

6. The requirements of 49 CFR 192 Subpart L—OPERATIONS are not addressed within the DEIS.

7. 30 CFR Part 380 states: "Describe measures used to exclude the public from hazardous areas. Discuss measures used to minimize problems arising from malfunctions and accidents **with estimates of probability of occurrence** (emphasis added) and identify standard procedures for protecting services and public safety during maintenance and breakdowns."

   There is no discussion within the DEIS as to how these requirements are addressed especially the probability and consequences of an accident and/or malfunction.

8. Based on the results of the Fukushima nuclear meltdowns the Social and Economic consequences may exceed $1 Trillion should an accident occur with consequential damage due to proximity to Indian Point and NYC. Consequential damages from secondary fires and explosions from the millions of gallons of fuel oil stored on the Indian Point site must also be considered

9. The Nuclear Regulatory Commission has specifically notified[8] all nuclear facilities of the potential dangers of locating gas lines in the vicinity of nuclear plants. Neither the Spectra application nor the DEIS address this major risk.

   There is no discussion of the potential for preventing terrorism and the impacts of such an event.

As stated in the DEIS: "To ensure that the AIM Project would not present new safety hazards to the IPEC facility, we are recommending that Algonquin file the final conclusions regarding any potential safety-related conflicts with the IPEC based on the Hazards Analysis performed by Entergy.

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[7] http://www.ecfr.gov/cgi-bin/text-idx?SID=feed3509ef9a6b39ee12360353228fd6&node=se49.3.192_1615&rgn=div8

[8] **Information Notice No. 91-63: Natural Gas Hazards at Fort St. Vrain Nuclear Generating Station**

It is imperative that this "Hazards Analysis" be performed by an independent, qualified party with oversight from representatives from local legislators and residents.

In summary, the proposed AIM project poses extreme dangers to the residents of Westchester County and surrounding areas that include pipe corrosion to the new and existing gas lines, damage due to installation and subsequent construction accidents, and other events that may impact the environment.

I would appreciate a detailed written response to these issues prior to the finalization of the DEIS.

Sincerely;

Paul M. Blanch
135 Hyde Rd.
West Hartford, CT 06117
860-236-0326

Cc: Chairman Allison M. Macfarlane
USNRC

Enclosure 4
12/17/2015 Letter to FERC Chairman

Paul M. Blanch
Energy Consultant

12/17/15

Chairman Norman Bay
Federal Energy Regulatory Commission
888 First Street N.E.
Washington, D.C. 20426

Benjamin.williams@FERC.gov

Re: Docket #CP14-96

Spectra Incremental Market (AIM) Project siting at the Indian Point Energy Center (IPEC)

Dear Chairman Bay,

The Spectra Algonquin Incremental (AIM) project is currently under construction and a tolling order has been in place since May 3, 2015. Many significant issues were raised in nine Requests for Rehearing, yet these issues remain unaddressed.

The final EIS issued by FERC on January 23, 2015 for the AIM Project states:

> *5.1.12 Reliability and Safety*
>
> *The pipeline and aboveground facilities associated with the AIM Project would be designed, constructed, operated, and maintained* ***in accordance with or to exceed the PHMSA Minimum Federal Safety Standards in 49 CFR 192 [emphasis added].*** *The regulations are intended to ensure adequate protection for the public and to prevent natural gas facility accidents and failures. The PHMSA specifies material selection and qualification; minimum design requirements; and protection of the pipeline from internal, external, and atmospheric corrosion.*

I have written a letter to the Department of Transportation, PHMSA, (copy enclosed) outlining many ways in which PHMSA has not assured compliance with 49 CFR 192. Specifically, 49

CFR 192.935 requires a risk assessment and 49 CFR 192.917 requires the operator to identify potential threats to pipeline integrity and use the threat identification in its integrity program.

49 CFR 192.935 clearly requires a risk assessment by stating:

> *§192.935 What additional preventive and mitigative measures must an operator take?*
>
> *General requirements. An operator must take additional measures beyond those already required by Part 192 to prevent a pipeline failure and to mitigate the consequences of a pipeline failure in a high consequence area. An operator must base the additional measures on the threats the operator has identified to each pipeline segment. (See §192.917) An operator must conduct, in accordance with one of the* ***risk assessment [emphasis added]*** *approaches in ASME/ANSI B31.8S (incorporated by reference, see §192.7), section 5, a* ***risk analysis of its pipeline to identify additional measures to protect the high consequence area and enhance public safety [emphasis added].*** *Such additional measures include, but are not limited to, installing Automatic Shut-off Valves or Remote Control Valves, installing computerized monitoring and leak detection systems, replacing pipe segments with pipe of heavier wall thickness, providing additional training to personnel on response procedures, conducting drills with local emergency responders and implementing additional inspection and maintenance programs.*

I urge you to rehear the decision to grant Spectra Energy Partners a Certificate of Public Convenience and Necessity and to institute a stay of construction due to the national security and safety issues related to the siting of the Spectra AIM pipeline at the Indian Point nuclear facility. The Federal Energy Regulatory Commission based its approval on the faulty Entergy risk assessment and the NRC's confirmatory analysis that does not properly evaluate the risk of a gas pipeline rupture and the potential for a catastrophic result in the densely populated New York City metropolitan region. FERC also failed to provide assurance that all of the requirements of 49 CFR 192 were being met.

Enclosed please find my comments regarding the Nuclear Regulatory Commission's responses to 39 questions I submitted on July 27, 2015. The NRC's letter of November 6, 2015 was sent long after the 4-6 weeks that the NRC estimated when Senator Gillibrand's aide inquired about the length of time it would take for the questions to be answered. A letter from Congressman Eliot Engel and Congresswoman Nita Lowey dated October 21, 2015 urged the NRC to send the responses within two weeks and the long-awaited responses finally arrived on November 6, 2015.

I have taken time to address each of the NRC's responses and I urge you to require PHMSA to provide a copy of its or Spectra's risk assessment and how all of the requirements of 49 CFR 192 are being met as stated in FERC's EIS

I have confirmation from the NRC that FERC's EIS, the AIM project is not in compliance with 49 CFR 192.615 and other requirements as outline in my enclosed letter to the PHMSA Administrator.

How can we be reassured that the AIM project is in compliance with 49 CFR 192, especially with the requirement for a risk assessment?

Please feel free to contact with me with any questions.

Sincerely,

Paul M. Blanch
135 Hyde Rd.
West Hartford, CT 06117
860-236-0326

McCloskey, Bridin

Subject: FW: Letter to the FERC Chairman
Attachments: 20180625 Final signed Letter to FERC Chairman.pdf

From: Paul [mailto:pmblanch@comcast.net]
Sent: Monday, June 25, 2018 11:11 AM
To: Amy Rosmarin <amyrosmarin@aol.com>; Susan Babdolden <svandolsen@gmail.com>
Cc: Paul Blanch <pmblanch@comcast.net>; Haagensen, Brian <Brian.Haagensen@nrc.gov>; Holian, Brian <Brian.Holian@nrc.gov>; Raspa, Rossana <Rossana.Raspa@nrc.gov>; Spicher, Terri <Terri.Spicher@nrc.gov>; Paul Gallay <pgallay@riverkeeper.org>; Richard Webster <rwebster@riverkeeper.org>; Maggie Coulter <mcoulter@riverkeeper.org>; Ellen Weininger <eewgrassroots@aol.com>; Ellen Weininger <eewgrassroots@gmail.com>; marilyn elie <eliewestcan@gmail.com>; John J Sipos <john.sipos@ag.ny.gov>; Tom Congton <Thomas.Congdon@dps.ny.gov>; Dave A Lochbaum <dlochbaum@ucsusa.org>; Criscione, Lawrence <Lawrence.Criscione@nrc.gov>; Kim Fraczek <kim@saneenergyproject.org>; Paula Clair <pclair3@aol.com>; Nancy Vann <nancy_vann@hotmail.com>; Nancy Vann <nancy.vann@gmail.com>; Courtney M. Williams <mazafratz@yahoo.com>; Tina Volz-Bongar <tina@bongarbiz.com>; Jessica Roff <jessroff@gmail.com>; Patrick Robbins <patrickopticon@gmail.com>; JK (Lilith) Canepa <jkthecat@gmail.com>; Shay O'Reilly <shay.g.oreilly@gmail.com>; Dorfman, Prof. David N. <ddorfman@law.pace.edu>
Subject: [External_Sender] Letter to the FERC Chairman

The enclosed letter was sent to and received by the FERC Chairman's office this AM.

If anyone is so inclined, this letter may be used as a basis for a letter to the New York State agencies. In my opinion the State letter was nothing more than "Cuomo's Red Herring" to absolve the State of any malfeasance. Cuomo has personally been aware of this issue since 2009/2010 when I identified it to his AG office during my time as a consultant to Cuomo's office. My contract was terminated shortly after this.

I would appreciate it if you would distribute this letter to any and all interested parties and organizations.

Begin forwarded message:

From: Paul <pmblanch@comcast.net>
Subject: Letter to the FERC Chairman
Date: June 25, 2018 at 10:45:08 AM EDT
To: CUSTOMER@FERC.GOV
Cc: Paul Blanch <pmblanch@comcast.net>, Amy Rosmarin <amyrosmarin@aol.com>, Susan Babdolden <svandolsen@gmail.com>

Please assure that the enclosed is delivered to the Chairman and his staff.

Paul Blanch
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West Hartford, CT 06117
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860-236-0326
Cell 860-922-3119