Law Offices of
Ernest C. Hadley, P.C.

1040 B MAIN STREET
P.O. BOX 549
WEST WAREHAM, MA 02576

ERNEST C. HADLEY
JOLEEN PAYEUR OLSEN

(508) 291-1354
FAX (508) 295-7002

OF COUNSEL
WAITE P. STUHL

March 3, 1997

Leonard J. Callan
Executive Director of Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

re: Request for Licensing Actions
10 C.F.R. § 2.206
Northeast Utilities
Millstone Units 1, 2 and 3, and
Connecticut Yankee

Dear Mr. Callan:

Pursuant to 10 C.F.R. § 2.206, this letter constitutes a petition filed on behalf of Albert A. Cizek, formerly a Senior Engineer and Engineering Supervisor at Northeast Utilities ("NU"), currently employed in NU's Employee Concerns Program ("ECP"), to institute a proceeding under 10 C.F.R. § 2.202, to modify the licenses issued to NU to operate the Millstone Nuclear Reactors, Unit 1, Unit 2 and Unit 3, and Connecticut Yankee by placing certain conditions, specified herein, on the operating licenses of each of those facilities. As grounds for this petition, Mr. Cizek maintains that NU has knowingly, willingly and recklessly operated Millstone Unit 1, Unit 2, Unit 3 at Waterford, CT, and its Connecticut Yankee Nuclear Power Plant at Haddam Neck, CT, in violation of their respective operating licenses, the regulations of the Nuclear Regulatory Commission ("NRC"), and their respective Updated Final Safety Analysis Reports ("UFSAR") for a prolonged period of time, which unnecessarily but significantly compromised public health and safety by eroding the required defense in depth philosophy; that NU has knowingly, willingly and intentionally harassed, intimidated and discriminated against its employees who raise safety concerns in violation of United States statutes and NRC regulations for a prolonged period of time, which also has unnecessarily but significantly compromised public health and safety by eroding the required defense in depth philosophy; and, that in the absence of express license conditions, there is no reasonable assurance that NU will cease and desist from engaging in these activities in the

9704110169 970408
PDR ADOCK 05000213
G PDR

future. The basis for the petition is set forth in further detail below.

Petitioner also requests that public hearings on the petition be scheduled in the immediate vicinity of the Millstone Nuclear Power Station and Connecticut Yankee Nuclear Power Station for the presentation of further evidence in support of the petition. Petitioner specifically requests that these public hearings be held and decision on this petition issued prior to the restart or decommissioning of any of these units.

Proposed License Conditions

Petitioner proposes the operating licenses of Millstone Units 1, 2 and 3 and Connecticut Yankee[1] each be modified to include the following provisions:

1. Within 30 calendar days of receiving a total of three license violations from the U.S. Nuclear Regulatory Commission during any three year period, irrespective of the violation level, the operating license of the facility shall be suspended for a period of not less than 90 days and not more than 180 days.

2. Within 30 calendar days of receiving a total of three violations of 10 C.F.R. Part 50, including all applicable appendices, from the U.S. Nuclear Regulatory Commission during any three year period, irrespective of the violation level, the operating license of the facility shall be suspended for a period of not less than 90 days and not more than 180 days.

3. Within 30 calendar days of receiving a total of three violations of the UFSAR from the U.S. Nuclear Regulatory Commission during any three year period, irrespective of the violation level, the operating license of the facility shall be suspended for a period of not less than 90 days and not more than 180 days.

4. Within 30 calendar days of receiving any harassment, intimidation and discrimination ("HI&D") finding by the U.S. Nuclear Regulatory Commission, the U.S. Department of Labor, or any state or federal court of competent jurisdiction, the operating license of the facility shall be suspended for a period of not less than 90 days and not more than 180 days.

---

[1]/ Although NU has made an economic decision not to restart Connecticut Yankee, licensed activities still are being conducted at that facility.

5. If, within five years of a license suspension based on paragraphs 1 through 4 above, the licensee receives a total of three license violations from the U.S. Nuclear Regulatory Commission, irrespective of the violation level; receives a total of three violations of 10 C.F.R, Part 50, including all applicable appendices, from the U.S. Nuclear Regulatory Commission, irrespective of violation level; receives a total of three violations of the UFSAR from the U.S. Nuclear Regulatory Commission, irrespective of violation level; or receives any HI&D finding by the U.S. Nuclear Regulatory Commission, the U.S. Department of Labor, or any state or federal court of competent jurisdiction, the operating license of that facility shall be permanently revoked within 90 calendar days.

6. In the event that the license of a facility is revoked pursuant to paragraph 5, no operation of that facility for the purpose of generating electric power shall be permitted during the pendency of any administrative or judicial processes or appeals related to such revocation.

7. In the event that the license of a facility is suspended or revoked under paragraphs one through five, the U.S. Nuclear Regulatory Commission shall designate an appropriate licensee to maintain the facility in shutdown mode for the duration of the suspension or until such time as a new licensee is found to operate the facility.[2] NU shall be responsible for all expenses related to the operation of the facility during such shutdown. NU shall be required to post a bond in the amount of $500,000,000 (five hundred million) as reasonable assurance that it can fulfill this requirement.

Petitioner further requests that these conditions be imposed on the operating licenses of Millstone Units 1, 2 and 3 prior to Commission approval to start-up any of those plants, and further requests that these conditions be imposed on the operating license of Connecticut Yankee prior to any decommissioning of that plant.

---

[2]/ Since even maintaining a facility in shutdown mode is a licensed activity, it will be necessary for the Commission to designate a licensee to maintain the facility during any such shutdown. The only other alternative is for the NRC to operate the facility—an alternative which is not desirable given the Commission's lax regulatory posture and which is of questionable legality.

Rationale for Conditions

Since August 1995 a series of public revelations about past operations of Millstone Units 1, 2 and 3 and Connecticut Yankee have eroded the public confidence in Northeast Utilities to safely and legally operate any or all of its nuclear power plants. Corresponding revelations about the abdication of regulatory responsibility by the NRC have similarly eroded the public confidence in the NRC to adequately protect the public health and safety.

The only proper conclusion to be drawn from these revelations is that NU is unfit to hold any license to operate a nuclear power plant. Toward that end, one group of Petitioners—Citizens Awareness Network ("CAN") and the Nuclear Information and Resource Service ("NIRS")—has filed a petition under 10 C.F.R. § 2.206 to revoke the operating licenses of all NU plants in Connecticut. However, reality dictates that the petition has no chance of success. Indeed, the NRC has yet to rule on the requested licensing action in a petition filed by George J. Galatis and We the People, Inc., of the United States in August 1995, despite the fact that every allegation in the petition has been proved true, has been verified by the NRC Office of Investigations or Office of Inspector General, and currently is the subject of a criminal probe by the United States Attorney for the District of Connecticut. Since there is virtually no chance the NRC will take the appropriate licensing action, the Petitioner requests that, in lieu of such action, the Commission impose certain self-executing conditions on the operating license of each facility.

From a practical perspective, the NRC has given the public only an illusory process in 10 C.F.R. § 2.206. Two of the permissible requested actions—revocation and suspension of a license—cannot be granted. Even to maintain a nuclear power plant in a shutdown mode requires the performance of licensed activities. The resulting reality is that the NRC cannot revoke or suspend an operating license because the licensee then would be unable to perform the licensed activities required to safely maintain the shutdown plant. Thus, an operating license can be revoked only if the NRC finds another qualified licensee to oversee operations during shutdown. That is a highly unlikely event given the fact that shutdown plants generate no revenues but conversely generate extremely high expenses.

Also, because the NRC has allowed one utility to construct and operate four nuclear power plants in such a small geographic area, revocation of NU's licenses would likely cause economic chaos in the State of Connecticut. Loss of jobs, the cost of replacement power and the ultimate cost of decommissioning would end up penalizing the residents of Connecticut, who have innocently been exposed to unnecessary health and safety risks, instead of NU and its officers and executives, who have engaged in deliberate and

willful misconduct.

NU has promised, time and time again, to address its failings without success. Although in different positions, most of the individuals responsible for these failings remain as part of NU management or staff. Executive NU management maintains its position that the past should be ignored in restart considerations. This position apparently is embraced by the NRC as evidenced by its willingness to commit virtually all of its resources to working with NU on its restart plans, and virtually none of its resources to punishing NU for its past transgressions. Moreover, NU is attempting its recovery with primary reliance on loaned, short-term management that may not be around to provide the required checks and balances once normal operations resume and in the long term. Common sense and logic dictate that there is no reasonable assurance that NU will cease and desist from violating its operating licenses, the regulations of the NRC and its UFSAR's. Nor will it cease and desist from harassing, intimidating and discriminating against its employees who raise safety concerns. Trust has been lost. Only a sustained period of operating within the parameters of the law, NRC regulations, license and UFSAR requirements, without deviation or set back, can restore that trust.

For these reasons, and because the regulator cannot be trusted to exercise good judgment on behalf of the public it is mandated to protect, the placement of precise limiting conditions, along with adequate financial precautions, on the operating licenses of Millstone Units 1, 2 and 3 and Connecticut Yankee is the only logical means of attempting to ensure the plants are operated safely and legally in the future. Thus, although the Petitioner firmly believes that NU has demonstrated that it is unfit to hold any operating licenses, Petitioner instead seeks modification of the licenses of Millstone Units 1, 2 and 3 and Connecticut Yankee as the only logical safeguard.[3]

Contentions of Petition

Petitioner seeks the limiting conditions on the operating licenses of Millstone Units 1, 2 and 3 and Connecticut Yankee based on the following contentions:

- *Contention No. 1:* NU has knowingly, willingly and recklessly operated Millstone Unit 1, Unit 2, Unit 3 at

[3]/ Petitioner does acknowledge that the ultimate effectiveness of the license conditions depends heavily upon proper regulation by the NRC. There is, in Petitioner's opinion, no concrete evidence that such regulation is forthcoming. However, the NRC's failure to adequately regulate is not properly the subject of a petition under 10 C.F.R. § 2.206.

Waterford, CT, and its Connecticut Yankee Nuclear Power Plant at Haddam Neck, CT, in violation of their respective operating licenses, the regulations of the NRC, and their respective UFSAR's for a prolonged period of time, which unnecessarily but significantly compromised public health and safety by eroding the required defense in depth philosophy.

o *Contention No. 2:* NU has knowingly, willingly and intentionally harassed, intimidated and discriminated against its employees who raise safety concerns in violation of United States statutes and NRC regulations for a prolonged period of time, which unnecessarily but significantly compromised public health and safety by eroding the required defense in depth philosophy.

o *Contention No. 3:* In the absence of express license conditions, there is no reasonable assurance that NU will cease and desist from engaging in these activities in the future.

Each of the contentions is addressed below.

Contention No. 1

The evidence is overwhelming that, for several years now, NU has operated as a "rogue utility" deliberately placing cost control over safety and intentionally violating the conditions of its operating licenses, NRC regulations and UFSAR's. For example, under 10 C.F.R § 50.73, NU, like all licensees, is required to report certain specified events in License Event Reports ("LER"). Most of the events required to be reported by the regulation specifically relate to license violations, regulatory violations, UFSAR violations, *e.g.*, operations outside of a plant's design basis, or violations of the principles of sound engineering judgment. By its own admissions through LER's, which have been incomplete, inaccurate, misleading and intentionally false, NU has an abysmal record. Over the last four years, the following number LER's have been filed for Millstone Units 1, 2 and 3 and Connecticut Yankee:

YEAR	UNIT	UNIT	UNIT	UNIT
	MP1	**MP2**	**MP3**	**CY**
1993	25	23	23	19
1994	33	43	15	29
1995	32	45	22	23
1996	65	41	50	30

The industry average for LER's per plant for 1993 through 1995 has been 13, 12 and 11, respectively. *See*, Annual Report, 1994-FY 95, Office for Analysis and Evaluation of Operational Data (July 1996) at p. 40.[4]

Moreover, NU, at all of its Connecticut plants, has violated license and regulatory requirements, cast aside good engineering judgment and intentionally operated the plants outside of their design bases. Revelations of this type of conduct in the past 18 months are too numerous to fully recount in this petition. However, because extensive documentation already exists of these examples, Petitioner requests that the NRC take administrative notice of certain documents listed in the section entitled, "Evidence in Support of Petition," *infra*.

But Petitioner's request for modification of the operating licenses of Millstone Units 1, 2 and 3 and Connecticut Yankee also is based on personal experience. In one matter in which Petitioner was directly involved as the alleger, NU willfully operated Millstone Unit 1 for over 20 years without testing, as required, certain containment isolation valves, including CU-29.

Petitioner originally challenged the decision of NU to place testing of CU-29 in the Integrated Safety Assessment Program ("ISAP"). ISAP, as Petitioner pointed out to NU, was designed to deal with new and emerging regulatory issues that had arisen since the licensing of the plant. Testing of CU-29 was a requirement by virtue of the NRC's adoption of 10 CFR Part 50, Appendix J in February 1973. Furthermore, it became a license condition when the full term operating license ("FTOL") was granted in 1986. Thus, it was clearly part of the licensing basis of Millstone Unit 1 and, as such, was improper for inclusion in ISAP. Further, Petitioner challenged the designation of testing of CU-29 as a "low priority" in ISAP.

When, because of Petitioner's continued insistence, the valve was tested in 1995 it was determined that the valve leaked excessively and could not perform its intended safety function. Significantly, in a report by the U.S. Nuclear Regulatory Commission Office of Inspector General ("OIG"), entitled "NRC STAFF ACTIONS TO ADDRESS CU-29 ISOLATION VALVE ISSUE," Case No. 96-06S, dated September 3, 1996, the OIG stated:

> The OIG learned that in February 1973, the NRC issued Appendix J "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors", which became a requirement for all licensed nuclear power reactors. Specifically, Appendix J requires that all operating licensees for water-cooled power reactors test the leak-

---

[4]/ No data is currently available for 1996.

tight integrity of the primary reactor containment, systems and components, including containment isolation valves. Appendix J tests are required to be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

In August 1975, NRC requested NU to determine whether Millstone Unit I was in full compliance with Appendix J and if not, to identify planned actions and to prepare a schedule to achieve compliance. NRC advised NU that possible courses of action included modifications to design features to permit conformance with the testing requirements as well as requests for exemptions from Appendix J requirements. In November 1975, in response to the NRC's request, NU provided a summary of containment isolation valves and identified tests conducted to that point. NU also identified valves which would require exemption from Appendix J requirements. Between 1975 and 1984, the NRC staff and NU exchanged correspondence regarding the status of NU's compliance with Appendix J.

In November 1984, NRC initiated the Integrated Safety Assessment Program (ISAP) to conduct integrated assessments for operating nuclear power reactors. The ISAP was intended to address plant-specific evaluations of licensing actions, plant improvements and unresolved generic safety issues. Millstone Unit I was one of two operating plants selected by the NRC to participate in the ISAP pilot program.

[I]n April 1988, NU requested exemptions relating to Appendix J for certain containment penetrations, including containment isolation check valve CU-29. NU requested an exemption from testing requirements for check valve CU-29 because design features of the check valve did not permit testing unless certain modifications were made.[5] In June 1991, the NRC denied this exemption request. Millstone Unit I was shut down for refueling outage 13 when NRC denied the exemption request; however, at the end of outage 13 the licensee resumed plant operations without testing valve CU-29.

Between October 1992 and June 1995, NRC reviewed and concurred on several of NU's ISAP reports which outlined plans to modify and test valve CU-29 in accordance with Appendix J during refueling outage 15. Ultimately, when

---

[5]/ What the Inspector General Report does not make clear is that NU twice requested exemptions for CU-29 and was twice turned down by the NRC.

tested during refueling outage 15, NU determined that valve CU-29 leaked excessively and may not have been capable of performing its intended containment isolation function.

Id. at pp. 4-5. Even more significant is the fact that CU-29 was designed to provide containment isolation in the event of a single active failure to CU-28, a motor-operated valve. It has been determined that CU-28 was "not environmentally qualified to perform its containment isolation under adverse conditions." *Id.* at p. 16. As such, this penetration and primary containment were outside of design basis, which compromised public health and safety by eroding the defense in depth philosophy.

Likewise, Unit 2, has seven containment isolation valves which lack environmental qualification.[6] *See*, Combined NRC Inspection Report Nos. 50-245/96-06, 50-336/96-06, 50-423/96-06. In a letter to Ted C. Feigenbaum, Executive Vice President and Chief Nuclear Officer, NU, dated October 9, 1996, accompanying the inspection report, Wayne D. Lanning, Director, Millstone Oversight Team wrote:

> The third apparent violation at Unit 2 involved the lack of environmental qualification for seven containment isolation valves that must be re-opened during the post-accident phase of an accident. This issue is of particular concern because qualifications of four of the seven valves were the subject of specific NRC review in 1988, and the safety function for these valves was not identified and corrected at that time. Further, the NRC identified other weaknesses in the implementation of the environmental qualification program that raise questions regarding the ability of EEQ components to perform their safety function. Therefore, NRC considers the completion of outstanding EEQ and high energy line break program activities, and the revalidation of the qualification of affected components to be a plant start-up issue.

Id. at p. 2. Thus, as with past infractions, problems which surface at one unit usually are indicative of problems that exist

---

6/ In an LER filed on April 25, 1996, NU reported that at Millstone Unit 2 the safety functional requirements ("SFR's") of the seven Electrical Equipment Qualification ("EEQ") solenoid operated valves ("SOV's") could not be demonstrated. The report noted the "SFR's indicate that these SOV's are required for containment isolation and subsequent post-accident operation." LER 96-019-00, April 25, 1996 at p. 1. The incident is safety significant because NU "cannot demonstrate reasonable assurance that the 7 listed EEQ valves would have performed their safety function on demand post-accident." *Id.* at p. 4.

at NU's other units.

In addition to evidence that problems that are identified at one unit also exist at other units, there is overwhelming evidence that multiple, related problems have been allowed to exist that compromise safety beyond any acceptable limits. For example, in conjunction with the problems previously outlined with regard to CU-29 and CU-28, piping system welds associated with those valves suffered from intergranular stress corrosion cracking ("IGSCC")—a phenomenon of which the industry has been aware for many years and which was the subject of NRC Generic Letter ("GL") 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."

The above-cited inspection report concluded:

> During the RFO 15 inspection, thirty five (35) welds were examined containing IGSCC. Fourteen of the 35 welds were previously identified by the UT[7] Level II technician, as early as 1984, as having at least one IGSCC indication, and were subsequently overturned by the NDE[8] Level III. The disposition of the 35 welds with IGSCC, during RFO 15, was to replace the component or weld overlay the components prior to start-up.
>
> Six reactor coolant components (RCAJ-2, RRJJ-4, RREJ-4, RRCJ-4 and CUBJ-18) with flaws were placed inservice, between 1984 and 1995, without flaw analysis as required by ASME Section XI, 1986 Edition, Paragraph IWB-3640. During the UT examinations, each component had at least one IGSCC indication. The ASME Section XI analysis was not performed on the components because the UT Level III inappropriately evaluated the IGSCC indications to be geometry. The indications were determined to be cracks during refueling outage 15. The licensee performed an evaluation during the November 1995 refueling outage, RFO 15, in accordance with ASME Section XI, 1986 Edition, IWB-3640, to determine the operability to the components. The licensee determined the components did not meet the requirements for continued service and declared the components inoperable. The licensee defined inoperability of a component as a decrease or elimination of the operating safety margin for structural integrity. The licensee determined the safety margin is decreased when a crack through wall dimension in the component is

[7]/ "Ultrasonic Testing."

[8]/ "Non-Destructive Examination." It should be noted that the NDE Level III is an industry certified person. The number of "errors" made by the NDE Level III in this case is strongly indicative that the "errors" were intentional.

> equal to or greater than 75% of the pipe wall nominal thickness.
>
> The six components had intergranular stress corrosion cracks (IGSCC) that were greater than 75% through the wall. Two of the six components leaked during preparation for weld overlay. The reactor coolant systems were degraded to the extent a detailed evaluation was necessary to determine system operability. The results of the licensees (sic) evaluation determined *the six components had an unacceptable structural integrity and a high probability of abnormal leakage.*

Id. at § M1.2b, ISI Program Review (emphasis added).

Of particular significance is the finding with respect to pipe weld CUBJ-18 because of its direct connection with CU-29. In Licensee Event Report ("LER") 95-029-00, dated December 15, 1995, NU stated:

> A structural review was performed for CUBJ-18 and concluded that operability may not have maintained in the event of a design basis seismic event. *In the worst case, and the weld had failed, leakage from the reactor would have been limited by check valve 1-CU-29. This valve was inspected during this outage and verified that it would have shut and prevented gross leakage from the penetration. . . .*

Id. at p. 4 (emphasis added). In other words, one of the pipe welds most susceptible in the event of a seismic event was in the worst possible location, *i.e.*, between CU-29 and CU-28, and outside primary containment.

Although NU, in LER 95-029-00, took credit for an operable CU-29 check valve, in LER 96-012-00, dated March 7, 1996, NU conceded:

> On December 3, 1995, with the plant shutdown and the reactor in COLD SHUTDOWN condition, it was determined that a reactor water cleanup system *containment isolation check valve, 1-CU-29, had exceeded its maximum rate leak while it was in operation. . . .*

Id. at p. 1 (emphasis added).[9] NU concluded that there "were no adverse safety consequences" as a result of CU-29 being inoperable "since the redundant valve in the penetration was capable of performing the containment isolation function." *Id.* at p. 2. In making this determination, NU deliberately and intentionally omitted any consideration of the IGSCC that had occurred to pipe

---

[9]/ Nowhere in LER 96-012-00 does NU mention its earlier findings regarding IGSCC and pipe weld CUBJ-18.

weld CUBJ-18 per LER 95-029-00, despite the fact that the design basis for primary containment includes a "design basis earthquake (DBE)" according to the UFSAR. *See*, UFSAR at Sec. 6.2.1.1.1.

Not only do the above-described events constitute a course of deliberate and intentional misconduct, as well as deliberate and intentional material false statements, but it is clear that NU, through these actions, significantly jeopardized the public health and safety. Far from the "defense in depth" strategy employed in the design and operation of nuclear power plants, NU allowed Millstone Unit 1 to operate in conditions where a single failure would have resulted in an uncontrolled release of radiation. Pipe weld CUBJ-18 had an excessively deep crack which made the piping system inoperable; simply stated, an earthquake could result in a failure of the piping system allowing a direct release path of reactor coolant or containment atmosphere outside primary containment through leaking CU-29 with no means to isolate.

Petitioner also has direct knowledge that the deliberate and intentional disregard for license requirements, NRC regulations, good engineering judgment and the design basis of the Millstone units continues in the present under NU's new management. For example, during the November 1995 refueling outage, Petitioner observed "scale" in the Low Pressure Coolant Injection ("LPCI") heat exchanger tubing after the tubing was hydrolazed.

Problems with scaling had been identified and addressed by the NRC in GL 89-13, "Service Water System Problems Affecting Safety Related Equipment," dated July 18, 1989, and GL 89-13, Supplement 1, dated April 4, 1990. According to Northeast Utilities Independent Root Cause Evaluation, A/R 96008360, "MP1 LPCI Heat Exchanger Tube Scaling," approved July 11, 1996:

> During 1989/1990, the NRC issued GL 89-13 and Supplement 1. The GL identified tubing scale (e.g., calcium carbonate deposits) as one type of fouling known to degrade service water heat exchanger performance. The plan outlined in GL 89-13 intended that the industry perform thermal performance testing of their safety related service water heat exchangers. The general response by the industry to GL 89-13, including NU, was to delay its implementation and/or pursue low cost alternatives to the thermal performance testing plan outlined by the NRC. . . .
>
> While NU was taking FRM[10] approach, the NRC was conducting Service Water System Operational Performance Inspections (SWSOPI's) throughout the nuclear power industry that included an assessment of utility responses

[10]/ "Frequent Regular Maintenance."

to GL 89-13. Early findings from these SWSOPI's were reported to the industry and highlighted to MP and CY Engineering Groups in memo ES-ME-94-076, dated February 18, 1994. This memo pointed out that the FRM approach was being seriously challenged by the NRC, basically because the NRC was finding an absence of a comprehensive, documented evaluation of each heat exchanger's capability to meet its design thermal requirements.

Id. at pp. 5-6.

During the November 1995 outage, Petitioner questioned the potential safety impact of the scaling on the heat exchangers, particularly given the low safety margin of the LPCI heat exchangers. Petitioner was told that thermal performance testing would be conducted prior to the end of the outage.[11] However, NU deliberately and intentionally took no action to perform such testing and, instead, prepared to restart without conducting the test. At that time (January 1996), Petitioner discussed the situation with the NU Employee Concerns Program ("ECP") which confirmed the low safety margin. However, ECP took no further action.

Finally, on March 26 1996, Petitioner initiated Adverse Condition Report ("ACR") 9801 declaring the LPCI heat exchangers inoperable. The Root Cause Analysis referenced above also found that Millstone Units 2 and Connecticut Yankee had no definitive inspection procedures for heat exchangers and that Millstone Unit 3 only had a visual inspection program to detect macrofouling by blue mussels. *Id.* at p. 6.

When it became apparent that NU still would not conduct the required test, Petitioner brought his concerns to the NRC, Region 1. On May 14, 1996, Richard Cooper, Director, Division of Reactor Projects, Region 1, referred Petitioner's allegation regarding LPCI back to NU through Ted C. Feigenbaum, Executive Vice President, Nuclear.[12] Although Mr. Feigenbaum was advised that the distribution of the allegation should be controlled and "limited to personnel with a 'need to know,'" the allegation received

---

[11]/ It is important to note that, at this time, Millstone Unit 1 was still in a planned refuel outage. That outage later was extended by a 10 C.F.R. § 50.54(f) letter and placement of Unit 1 on the NRC "Watch List." The plant was later degraded to a Category 3 plant and may not restart until given approval by a vote of the Commission.

[12]/ The referral was contrary to the NRC Volume 8, Licensee Oversight Programs, Management of Allegations Handbook. The NRC's referral of the allegation back to NU currently is the subject of an investigation by the Office of Inspector General.

widespread distribution at the Millstone site. This fact was acknowledged in the Independent Root Cause Evaluation cited above, which stated "[b]ecause of the uncontrolled distribution of the NRC transmittal, personnel involved with the LPCI scale issue appeared guarded and, at times, focused more on the NRC allegation than on the scale issue and any generic implications." *Id.* at Addendum, p. 1. One of the personnel involved with the Independent Root Cause Evaluation specifically referred to the matter as the "Cizek Affair."

The same Root Cause Evaluation recommends that "future NRC allegation issues be evaluated by a third party until NU . . . has the forthrightness to handle such issues appropriately." *Id.* at Addendum, p. 1. After continued persistence, Petitioner convinced the ECP Director to return "referred allegations" back to the NRC. However, the director's decision was overturned by Mr. Feigenbaum. Subsequently, at Mr. Feigenbaum's direction, "referred allegations" are addressed as standard practice within Nuclear Oversight without the involvement of ECP.

According to a subsequent root cause analysis performed by Millstone Unit 1 Engineering, "the reason for not assessing the affect on heat exchanger performance is . . . the lack of having comprehensive heat exchanger testing, monitoring, and visual inspection programs with clearly defined acceptance criteria." *See*, "Northeast Utilities Root Cause Investigation: Heat Exchanger Tube Side Scale Formation," Rev. 2, at p. 2, dated December 12, 1996. Typical of NU's intentional disregard of safety issues is the conclusion that the root cause was the failure to have an effective program. The root cause analysis fails to ask the obvious question: Why wasn't there such an effective program? The answer: Because NU had deliberately delayed developing and implementing such a program even after being advised by the NRC that the FRM approach was questionable, if not completely inadequate.

Other examples of NU's deliberate misconduct abound. For example, in Notice of Violation (NRC Combined Inspection Report No. 50-245/95-31; 50-336/95-31; 50-423/95-31), dated December 7, 1995, the Commission decided to excuse NU's conduct for two significant violations at Millstone Unit 1.

> The first violation involved an existing single failure vulnerability in the loss of normal power logic that would have prevented both emergency power sources from properly starting and sequencing required loads. This constitutes a violation of your technical specification because in the event of a single failure, following a loss of normal power during a Loss of Coolant Accident (LOCA), a loss of both emergency power sources would occur. The violation was caused by your inadequate design reviews of two different modifications installed

> in 1976 and 1989, which failed to detect the single failure vulnerability.
>
> . . . .
>
> The second violation . . . involved two examples of existing vulnerabilities in the standby gas treatment system (SGTS). In the first example, the system could rupture if a LOCA occurred while venting the drywell, which would result in a complete loss of SGTS because the drywell isolation valves would not close in time to prevent the pressure wave generated from the LOCA from affecting the integrity of both trains of the system's filters. Second, *an existing single failure vulnerability in the SGTS would prevent the system from mitigating the effects of a LOCA*. Specifically, during a LOCA, the inability to isolate one train on the SGTS, if it were to fail in conjunction with the inadequate backdraft damper design, would allow a short cycle flow path to be established. This flowpath would prevent establishing or maintaining the reactor building negative pressure, which is required to prevent a post LOCA ground level release.

Id. at pp. 1-2 (emphasis added). The violations regarding the SGTS are particularly significant since that is one of the major systems that would have been relied upon in the event of a release outside primary containment due to the previously cited CU-29 scenario. In short, NU operated Millstone Unit 1 in such a way that fundamental design basis assumptions could result in an unacceptable post-LOCA ground level release. Simply stated, multiple systems were outside their design basis which reduced safety margin and compromised public health and safety. Fortunately, no design basis accident occurred.

In an inspection between November 5 and December 26, 1995, NRC resident inspectors found other violations that are indicative of intentional misconduct, or at the least reckless disregard for license requirements. For example, at Millstone Unit 1, during shutdown:

> The operations staff failed to prevent work that had the potential for draining the reactor vessel while fuel removal was in progress, as required by technical specifications. The licensee does not have an adequate process to ensure all applicable technical specifications are implemented during refueling. *A violation was cited because it is an example of continuing problems with inadequate implementation of current regulatory requirements.* The licensee's subsequent position that

> established a 50 GPM[13] leak as threshold for the potential for draining the reactor vessel was inconsistent with the licensing basis that established the applicable technical specification.
>
> Operator actions to raise reactor pressure above the initial pressure assumed in accident analysis reflected a lack of understanding and respect for the plant's design basis. . . . [T]his issue is significant because *operators purposefully changed the normal operating practice without implementing appropriate procedural controls including the necessary safety evaluation*. In addition, the licensee failed to implement prompt procedure changes when the lack of procedural control over operating reactor pressure was identified on two prior occasions.

Executive Summary, Millstone Nuclear Power Station, Combined Inspection 245/95-42, 336/95-42, 423/95-42, February 6, 1996 at p. ii (emphasis added).

The same inspection report found that, during the November 1995 shutdown of Unit 3 "[a]n unplanned dilution of the reactor coolant system occurred due to inadequate procedures, poor communications between the Chemistry and Operations Department, and poor maintenance of the letdown high temperature divert valve." The report noted that this was the fourth unplanned dilution event at Unit 3 since December of 1993. *Id.* at p. iii.

In two other instances at Unit 3, the inspectors found:

> At Unit 3, licensee management displayed a lack of conservative safety perspective in not validating the conditions that exited with a leaking 10-inch unisolable check valve in RCS. The associated operability determination was based on engineering judgment and assumptions, and was not entirely accurate. . . .
>
> [T]he socket weld failures should have been prevented by corrective actions for similar vibration induced failures identified in 1994. The licensee postponed these socket weld repairs during the 1995 refueling outage, based in the inappropriate safety judgment that any socket weld failures would only result in a small RCS leak.

Id. at p v.

A December 13, 1995 LER provides another example of NU deliberately deciding to tolerate conditions that result in license violations.

---

[13]/ "Gallons per minute."

This time, the event involved Millstone Unit 2 where reactor core power exceed the rated thermal power of 2,700 megawatts for a period of some 11 hours. According to the LER:

> On November 15, 1995, at 1045 hours with the plant operating in Mode 1 at 100% power, a review of plant operating parameters identified that the reactor core thermal power level had inadvertently exceeded the maximum power level permitted by the operating license. This event was caused by incorrect steam generator blowdown flow rate value in the core heat balance calculation. This caused the calculated core thermal power to be less than the actual core thermal power. . . .

LER 95-043-00, December 13, 1995. As part of its solution to the problem, NU proposed simply that "the installation of process instrumentation to measure the blowdown rate is being evaluated . . . ." *Id.* The root cause of this event was a combination of inadequate design control, inadequate operator training and inferior, cost conscious corrective action—all of which are brought about by NU's intentional and deliberate disregard for the Millstone Unit 2 license and UFSAR and NRC regulations.

Other safety significant issues involving the determination of actual core power occurred at Millstone Unit 1 and Connecticut Yankee. Both involve the measurement of feedwater flow which is a key input into the calorimetric calculation which determines actual core power. In accordance with the operating licenses and technical specifications, actual core power must not exceed specified values to maintain the validity of assumed accident analyses described in the UFSAR's.

Shortly after the initial start-up of Millstone Unit 1 in December 1970, problems with the measurement of feedwater flow were encountered due to erratic and inconsistent differential pressure measurements from the primary flow elements—in this case Venturi tubes. As a short term, but questionable, solution, turbine first stage pressure, in conjunction with its corresponding flow coefficient and other parameters, was used to approximate steam flow.

This is not the preferred method to measure primary flow for use as an input into the calorimetric calculation for a variety of reasons.[14] These reasons drive the calorimetric calculation in

---

[14]/ Most important, knowledge of the moisture content of the steam is required. This is a difficult parameter to measure, so the unverified design or calculated value was used, although this was subject to a large error. Additionally, steam flow was used subject to unmeasured or unknown losses upstream of the turbine

a non-conservative, unquantified direction as components age and degrade. Regardless, a rigorous uncertainty analysis to bound core power error and assess the impact on assumed accident analyses described in the UFSAR was not performed.

The measurement was used well into the 1980's, when feedwater correction factors as large as six percent were applied to "adjust" measured feedwater flow. At this time, Petitioner, then an established engineering supervisor, and his immediate manager, discussed the subject and all of its inherent weaknesses with Millstone Unit 1 Engineering, but were rebuffed. Not until Petitioner conveyed the results of turbine cycle heat balance calculations, which suggested a net core power and corresponding electrical generation increase, did Millstone Unit 1 opt to address the problem. After replacement of the feedwater Venturis, core power decreased about one percent, meaning that past operation was in excess of licensed power. The change *could* have been much larger and more significant.

Sometime prior to 1986, Connecticut Yankee encountered problems with the measurement of feedwater flow by its primary flow elements—in this case orificed flow sections. Ultimately, one of the four orifice plates had to be replaced. However, Connecticut Yankee chose to manufacture its own orifice plate and install it without calibration. All Quality Assurance ("QA") requirements were ignored. Upon discovery by Petitioner's subordinates, Connecticut Yankee had no choice but to take action. During a subsequent refueling outage, all four orificed flow sections were replaced. Quite remarkably, comparison of the calibration results from an accredited flow laboratory showed only a small difference. However, the difference *could* have been quite large and significant. Connecticut Yankee essentially ran "blind" for an unknown period of time since a rigorous uncertainty analysis to bound core power error and assess the impact on assumed accident analyses described in the UFSAR could not be performed.

In both cases described above, NU demonstrated intentional and deliberate disregard for the Millstone Unit 1 and Connecticut Yankee licenses and UFSAR's, and NRC regulations. At Millstone Unit 1, this method was used in excess of 15 years. At Connecticut Yankee, the duration is unknown.

In an LER filed on December 13, 1996, NU reported at Millstone Unit 3, the "plant had operated in a condition that was outside the design basis due to a deficiency in specific design conditions for

---

which could be significant, primarily due to turbine bypass valve leakage to the condenser. Also, the turbine first stage flow coefficient is subject to change as the first stage geometries enlarge due to degradation, primarily due to erosion which is proportional to moisture content.

a system needed to remove residual heat and mitigate the consequences of an accident. It was determined that the Containment Recirculation System (RSS) spray piping and supports were not adequately designed for thermal loads resulting from accident temperatures." LER 96-007-02, December 13, 1996 at p. 1. This problem existed as part of the original plant design. *Id.* at p. 3. Again, the condition was significant "in that had the plant experienced a design basis accident in containment such as a LOCA or HELB, then the potential existed that these systems may not have been able to fulfill their required safety function." *Id.* at p. 4. Moreover, NU had to acknowledge that there were two similar events as reflected in LER 96-006-00, "Plant Shutdown Required by Technical Specifications for Auxiliary Feedwater Containment Isolation Valves Declared Inoperable" and LER 94-006-00, "Auxiliary Feedwater Pipe Restraints, Inadequate Design Due to Design Error." *Id.* at p. 5. Likewise, both of these deficiencies were original design deficiencies.

Also, at Unit 3, NU's intentional and deliberate decisions to ignore problems resulted in repeated leaks in four Reactor Coolant System ("RCS") loops through vibrational fatigue of small bore piping. This generally refers to a smaller diameter pipe (branch piping) welded to a socket (receiver,) which is, in turn, welded unsupported to a larger diameter pipe (main piping). Vibrational fatigue occurs when a stimulus, such as a running pump, vibrates the smaller diameter pipe and ultimately, through cyclic fatigue the smaller diameter pipe to socket weld will fail resulting in leakage.

At Millstone Unit 3, a small diameter pipe connected to the four RCS loops is used to locally measure pressure at sixteen locations. One weld failure resulting in leakage occurred May 1992. Two indications (faults) without leakage and one weld failure resulting in leakage occurred September 1994. One weld failure resulting in leakage occurred December 1995.

In September 1993, vibration testing was conducted to determine failure susceptibility. *See*, Memo from G.E. Dreschler, Component Test Services, to M.D. Hess, re: "Millstone Unit 3 - Reactor Coolant Piping Vibration Testing Flow Instrumentation Lines." The results showed negligible apparent susceptibility due to the lack of measured excessive vibration amplitudes. The report concluded that no further testing was necessary.

However, as noted, another failure occurred in September 1994. Consequently, a non-destructive examination was performed to identify faults as a precursor to failure. Two faults were identified. Unfortunately, existing non-destructive examination was not foolproof and all faults may not have been detected. Consequently, further review was required.

Additional examination of the September 1994 failure by an outside

source, ABB Combustion Engineering Nuclear Operations, confirmed cyclic fatigue. *See*, "Examination of the Socket Weld Crack in Millstone Unit-3 Loop C Pressure Tap V125," Draft Report, MISC-PENG-TR-028, December 1994. NU deleted two key points in the final version.[15] First, the final report deleted a notation in the draft that indicated:

> The US Nuclear Regulatory Commission (NRC) expressed their concerns to NUSCO about the cause of the failure.

Id. at p. 1-2. But more important, the following conclusion was deleted from the final report:

> Because the failure of one of these welds could result in a significant safety problem, it is recommended that each of the 16 similar welds be visually inspected every outage for the presence of boric acid deposits. Radiographing the welds has proven to be difficult and may not yield correct results. A review of any maintenance performed in the vicinity of the elbows may provide some insight into the cause of the failure. It is recommended that the vibration of the lines be monitored for a period of several days during normal operation of the plant.

Id. at p. 5-2.

Based on the above evolving scenario which demonstrated continued vulnerability, the Welding and Materials Engineering Section "strongly recommends that MP3 thoroughly review the existing piping design of the RCS instrument lines. Piping design modifications to eliminate socket welds or additional support modifications may be necessary to eliminate fatigue loading." Memo from A.J. Silvia, Component Engineering Services to George Pittman, MP3 Engineering Director, March 17, 1995, "MP3 Reactor Coolant System - Socket Weld Failure (3RSC*V125)".

Instead, NU management defended a past operability determination indicating the problem was of low safety significance since identification of socket weld failures is not unusual in the nuclear industry, socket weld failures are the most common weld failure, use of socket welds have been restricted to small lines, socket weld failures have demonstrated leak before failure, unidentified reactor coolant leakage is monitored by a system capable of detecting a 1 gpm change, there is a Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring System to monitor RCS leakage, any leakage would be collected within the containment drain system, the 3/4 inch diameter instrument line is limited by a 3/8 inch diameter port size, and normal makeup has

---

15/ The final version of the report is dated February 1995.

sufficient capacity to maintain pressurizer level and compensate for the problem. *See*, Memo from D.C. Gerber, Manager, MP3 Technical Support to M. Brothers, Director, Millstone Unit 3, March 22, 1995, "RCS Socket Weld Failure".

Subsequently, Mr. Pittman stated the present condition was acceptable to start-up from refueling outage ("RFO") 5 and support the upcoming operating cycle since an acceptable level of confidence existed not to incorporate proposed fixes during RFO 5. *See*, Memo from G.R. Pittman to M.H. Brothers, May 28, 1995, "RCS Elbow Tap Socket Weld Failures -- Current Status and Near and Long Term Corrective Strategies". Furthermore, according to Mr. Pittman, no safety issue existed due to the leak limiting feature of the configuration whereby complete severance could easily be made up by the charging pumps (RCS makeup). Additionally, Mr. Pittman stated there was minimal risk to lost generation. Basically, MP3 was willing to live with the problem and put off resolving it for at least one operating cycle even though the potential for a small break loss of coolant accident ("SBLOCA") existed due to a questionable design.

During the upcoming operating cycle, another failure occurred and was identified during a containment entry to identify RCS leakage for other suspected causes. Ultimately all sixteen locations were modified from the existing design to the current design.

Recent publications validated the shortsightedness of the decision to defer modification by stating the "ASME Code procedure does not ensure the design to be on the conservative side. . . . Moreover, either establishment of a new design code or confirmation of the authenticity of the current practice, as extended to higher orders of fatigue cycles, appears to be an urgent need." *See*, "Pressure Vessels and Piping Codes and Standards," Vol. 1, PVP-Vol. 338 at p. 3. But, the bottomline is that this is yet another example of intentional conduct on the part of NU to ignore existing safety problems, put off needed analysis and repairs, only to have the problem resurface under actual operating conditions.

As previously noted, this is precisely the same conclusion reached in the Executive Summary, Millstone Nuclear Power Station, Combined Inspection 245/95-42, 336/95-42, 423/95-42, February 6, 1996:

> [T]he socket weld failures should have been prevented by corrective actions for similar vibration induced failures identified in 1994. The licensee postponed these socket weld repairs during the 1995 refueling outage, based in the inappropriate safety judgment that any socket weld failures would only result in a small RCS leak.

Id. at p. v.

Since being forced to heighten inspections, by public disclosures

of NU's intentional misconduct and the NRC's prior knowledge of that misconduct, many of the NRC's own inspection reports have validated the contention that NU cannot and will not operate within the confines of its licenses, UFSAR's, and Commission regulations. For example, in a letter to Mr. Feigenbaum, dated July 31, 1996, regarding NRC Inspection Report 50-213/96-201, entitled "Special Inspection of Engineering and Licensing Activities at Haddam Neck," the NRC concluded:

> The [inspection] team found a number of significant deficiencies in the engineering calculations and analyses relied upon to ensure the adequacy of the design of key safety systems at Haddam Neck. In some cases, design-basis calculations and analyses were not sufficient to confirm that the safety system functional requirements would be met. Some of these errors were longstanding, . . . *These deficiencies revealed significant weaknesses in the defense-in-depth principles that the NRC relies upon to ensure that nuclear power plant operation does not jeopardize the health and safety of the public.* The team concluded that weaknesses in your configuration management processes and a lack of technical rigor, thoroughness, and attention to detail in the design process, either contributed to or directly caused the identified errors. . . .

Id. at p. 1 (emphasis added). The inspection team also found instances where commitments made to the NRC had not been met by NU. *Id.*

Summarizing the report, the NRC noted that NU's failure to act had larger implications for Connecticut Yankee:

> . . . The team found several instances involving the failure to identify, evaluate, and correct conditions adverse to quality, and some instances in which planned corrective actions were not promptly initiated. *In some instances, the delays in initiating planned corrective actions were significant because the actions included the evaluation of the potential generic implications of these issues for other plant systems and equipment.*

Id. at p. 2 (emphasis added).

Despite the fact that NU officials vehemently denied any correlation between its actions at the Millstone units, and indeed pointed to Connecticut Yankee as proof of its commitment to nuclear safety:

> The team found process issues at Haddam Neck which are similar to some of those identified at Millstone 1, as documented in the Event Response Team Report, dated

> February 22, 1996, that is commonly referred to as ACR 7007. As discussed above, *the team found that calculations did not exist to support some of the design-bases and administrative control programs at Haddam Neck have not maintained an accurate UFSAR.* In addition, licensee management oversight did not identify and address the patterns of corrective action program implementation problems, as discussed above. Finally, ACR 7007 states that a "general lack of understanding and appreciation for the relationship between 10 CFR 50, design-bases, licensing-bases, industry codes, and NU's administrative programs" existed. The team observed instances that demonstrated a similar lack of understanding by the licensee's staff at Haddam Neck.

Id. (emphasis added).[16]

The inspection report itself concluded:

> The inspection record for the last 2 years contains instances of problems which indicated weaknesses in the engineering programs and the modification processes. Instances included cases where inadequate designs were installed, engineering analyses were incorrect, or incorrect design-basis assumptions were used. For example, the incorrect design of an auxiliary feedwater flow instrument modification in 1995 resulted in the instruments being over-ranged when the main feedwater pumps were running (IR 95-19). In other examples, nonconservative LPSI flow assumptions were used in LOCA analysis (IR 95-27), nonconservative assumptions for metal mass were used in the containment analysis (IR 95-06), and nonconservative assumptions for the nuclear instrumentation detector response were used in the boron dilution analysis (IRs 94-09, 94-14). Although common causes were not identified, these issues indicated a weakness in the processes to verify accurate inputs to the licensing-basis analyses.
>
> The inspection report also contains examples where the licensing- or design-basis requirements for some systems were not maintained. Examples included the failure to meet the design codes for ECCS piping flanges (IR 94-03) and an inadequate seismic interaction analysis for reactor coolant support systems (IR 94-17).

Inspection Report at p. 53. The report went on to observe:

---

[16]/ Amazingly, although the report noted that the NRC was considering escalated enforcement action, not a single notice of violation has been issued by the NRC at this time.

> The most significant issue noted by the team was the failure of the licensee to appropriately consider design-basis scenario loads on the Class IE station batteries sizing calculations. Specifically, the licensee's calculations did not account for all of the loads associated with a LOCA coincident with loss-of-offsite power, and did not demonstrate that the battery voltage would remain above the minimum level required for operation of equipment. The licensee's evaluation resulted in the station batteries being declared inoperable and major load modifications being subsequently made to reduce loading on the batteries. Other deficiencies of particular significance identified by the team included failure of the licensee to adequately evaluate the effects of two-phase SW [service water] flow at the discharge of the CAR [containment air recirculation] fan coils during accident conditions, errors in engineering analyses and calculations supporting a proposed TS [technical specification] amendment for lowering CAR air flow, and errors in engineering analyses and calculations for the RWST level instrumentation.

Id. at p. 54.

Other NRC inspection reports support Petitioner's contention that NU's conduct was deliberate and intentional. Common sense simply dictates that the number of instances of desigh and licensing-bases discrepancies could not be due to mere negligence alone. For example, in a letter to Mr. Feigenbaum accompanying NRC Integrated Inspection Report 50-213/96-08, dated September 12, 1996, Mr. Cooper wrote that NU was expending considerable resources at Connecticut Yankee to addresses such discrepancies "needed to assure the continued operability of components important to safe plant operation." *Id.* at p. 1. For example, Mr. Cooper noted:

> The discovery by design engineering that the service water piping supplying cooling water to the CAR fans would not remain functional under accident conditions was an example of an issue for which the design basis for the plant had not been thoroughly reviewed or understood. Other design basis issues discussed in the enclosed inspection report included the reliance of high containment back pressure to assure reliable performance of the residual heat removal (RHR) systems under postulated accident conditions, and the adequacy of the containment sump screens to limit debris from entering the safety systems. *These issues impacted the operability of the emergency core cooling systems from performing their intended safety functions for certain postulated design basis events.*

Id. (emphasis added). Once again, the NRC noted five apparent violations that were being considered for "escalated enforcement" action, but no such action has taken place and no notices of violation have been issued.

The Executive Summary to the inspection report notes a number of instances of inadequate engineering support at Connecticut Yankee—a theme which appears repeatedly in violations at the Millstone units. For example, the Summary notes:

> [E]ngineering support to operations was inadequate in the failure to provide uncertainty calculations for the wide range nuclear instrumentation trip setpoint, and the failure of the plant design basis to adequately consider two phase flow conditions in the service water supply to the CAR fans. The latter deficiency, in combination with other design basis assumptions to assure adequate net positive suction head (NPSH) conditions for safety related pumps, *resulted in past plant operation with a potential for loss of containment heat removal, containment integrity and long term core cooling under postulated accident conditions.*
>
> . . . .
>
> Engineering support was inadequate to assure the emergency core cooling system (ECCS) flow path remained operable for all design basis conditions. Discrepancies included inadequate configuration management of the containment sump design and as-built conditions; a lack of detailed analysis and technical justification for the reliance on post-accident back pressure inside the containment to assure adequate NPSH for the residual heat removal (RHR) pumps, inadequate inspection/verification of sump as-built and material conditions; and, the lack of aggressive action in response to generic communications of industry events, which contributed to an inadequate operability determination regarding the sump screen design and mesh size.
>
> Poor engineering support was also noted to assure adequate controls exist for containment isolation valves that might not close in response to an isolation signal, and based on discrepancies in the design basis documentation for the core deluge valves. . . .

Id. at pp. 3-4 (emphasis added).

The problems relating to the containment sump pump design serve as a graphic example of the fact that NU's conduct was intentional and deliberate. As the inspection report noted, since the early 1980's the NRC issued several generic communications to licensees

"focus[ing] on the design and potential vulnerabilities of the containment sump." Inspection Report at p. 19. On November 18, 1980, the NRC requested that NU provide information on the design features and dimensions of the containment sump. Later, on December 3, 1985, the NRC issued Generic Letter 85-22 indicating it would not require a generic backfit to correct the problems. However, in 1986, NU reported that it had evaluated containment sump performance in its ISAP and considered the matter closed since the emergency sump was of standard design. *Id.* at pp. 19-20.

But:

> On July 26, 1996, the inspector observed licensee engineers perform a verification of the containment sump screen mesh size. The sump screen mesh according to in-plant walk down documentation in 1990 indicated that the mesh size had approximately 0.375 inch openings. The walk down on July 26, 1996 concluded that the mesh size was generally 0.50 inch, with a 3 inch by 2 foot gap on one end.

Id. at p. 19. The NRC inspectors concluded:

> The safety consequences of improper mesh in the containment sump is a potential failure to mitigate a loss of coolant accident due to debris clogging components within the emergency core cooling systems during long-term recirculation phase. The ability of operators to transition from injection phase to long-term recirculation phase is considered risk significant from the licensee's individual plant examination (IPE) report. The vulnerability accounts for approximately 16 percent of the analyzed total core damage frequency. . . .

Id. at pp. 21-22. The inspectors found the discrepancy had "significant safety consequences based upon plant risk and the ability to mitigate a postulated design basis accident." *Id.*

In late August and early September 1996, NU's willful and intentional disregard of UFSAR, licensing and regulatory requirements over an extended period of time led plant operators to place the Connecticut Yankee reactor in a position that nearly resulted in the inability to maintain adequate decay heat removal to ensure the public health and safety. The bottom line of this event is that NU continues to pose a significant risk to the public health and safety whether or not it operates its existing plants. The event seriously calls into the question the ability of NU to maintain safe conditions even when its plants are in shutdown mode. In a letter to Mr. Feigenbaum that accompanied NRC Augmented Inspection Team Review of Undetected Introduction of Nitrogen Gas into the Reactor Vessel during Plant Shutdown, Report No. 50-213/96-08, Region I Administrator Hubert J. Miller stated:

For approximately four days, control room operators were unaware that nitrogen gas was leaking into the reactor vessel and causing level to decrease. By September 1, 1996, reactor vessel level had decreased to approximately 3 feet below the reactor vessel flange. The decrease in reactor vessel level was potentially significant because a further decrease in level could have challenged the function of the operating heat decay removal system. While there were not actual public health and safety consequences of this event and adequate decay heat removal was maintained, the situation involving an unintended decrease in reactor water level in combination with the unavailability of decay heat removal equipment was safety significant.

. . . Several operations procedures failed to provide adequate details or contained incorrect information. The absence of acceptable procedures was a contributing cause for both the nitrogen gas intrusion going undetected and for the inadvertent diversion of water from the reactor coolant system (RCS). Several of the events were exacerbated by plant operators failing to follow plant procedures, conducting activities without procedural guidance, or making inappropriate decisions. A lack of a questioning attitude resulted in the failure to promptly identify the nitrogen gas accumulation in the reactor vessel. The failure by more senior operators to convey expectations to less experienced field operators during pre-job briefings resulted in inappropriate equipment manipulation that either directly caused or contributed to these events.

Id. at p. 1. The inspection report further found that the "poor material condition of several isolation valves," and past management failures to address previously raised concerns about vent header design contributed to the event. *Id.* at p. 2. Contributing factors also included inadequate procedures, failure to implement procedures, lack of a questioning attitude, inappropriate decision-making, inadequate pre-job briefings, failure to report the event, failure to conduct planned training, avoidable delays due to the RHR pump repair, poor plant material condition, untimely technical response, poor implementation of technical information, weak engineering/operations interface, inappropriate outage scheduling decisions, lack of direct reactor vessel indication and slow initiation of event response team. Inspection Report at pp. 10-16. In other words, the event was caused by NU's deliberate and intentional disregard of numerous requirements imposed by Connecticut Yankee's license and UFSAR, technical specifications, and Commission regulations. Still, no notice of violation has been issued by the NRC.

Yet, on November 2, 1996, at Connecticut Yankee an event that

occurred in the fuel transfer canal and reactor cavity led to the unnecessary exposure to workers to airborne radioactive material. The NRC identified five "apparent" violations, "some with multiple examples of non-compliance, . . ." Letter to Mr. Feigenbaum accompanying NRC Inspection Report 50-213/96-12 at p. 2. Included in the findings of the Inspection Report are the following incidents which demonstrate not only intentional and deliberate misconduct, but a callous disregard for the health and safety of its employees:[17]

> On November 2, 1996, two workers in the fuel transfer canal unknowingly collected, handled, and transported radioactive material (debris) with contact radiation levels ranging from 20 R/hr to 60 R/hr. The debris was not surveyed as it was collected, handled or transported. Such surveys were necessary and reasonable to ensure conformance with the occupational dose limits.
>
> On November 2, 1996, airborne radioactivity surveys were not adequate to detect high concentrations of airborne radioactivity within the fuel transfer canal as workers collected highly radioactive dry dirt like debris therein. Such surveys were reasonable in that areas traversed and worked in by the workers exhibited loose surface contamination level measuring up to 80 mrad/hr (beta) contamination and up to 30,000 disintegrations per minute/100 square centimeters alpha contamination (dpm/100 cm2).
>
> On November 2, 1996, airborne radioactivity surveys were not adequate to detect high concentrations of airborne radioactivity within the reactor cavity to support reactor stud hole cleaning. As a result, two workers were permitted to enter the reactor cavity notwithstanding the presence of high levels of airborne radioactivity.
>
> As of November 7, 1996, the licensee had not effectively evaluated the potential exposure of two workers, known to have been exposed to high levels of airborne radioactivity, sufficient to make the determination that the workers had substantial potential to exceed

---

17/ This event is remarkable, too, in that it exhibits the same willingness to deliberately jeopardize the health and safety of its workers, that NU exhibited in its refueling practices at Millstone Unit 1. At Unit 1, NU sent workers in to begin preparations for moving fuel well in excess of the required hold time. However, such action did not stop NU from publicly arguing that its full core offload practice at Unit 1 was safer for its workers.

> applicable regulatory limits relative to intake of alpha emitting isotobes on November 2, 1996.

Inspection Report at p. iii. The report goes on to note numerous other violations in connection with the incident.

In the letter to Mr. Feigenbaum, accompanying the report, the NRC noted:

> The NRC inspection identified significant deficiencies in the oversight and control of licensed activities, including programmatic breakdown in radiological controls and poor work planning, control, and practices relative to defueling activities on November 2, 1996. As a result, personnel were exposed to high concentrations of airborne radioactive material and handled highly radioactive debris, resulting in a substantial potential for an occupational exposure in excess of NRC regulatory limits. We are particularly concerned about your organization's failure to (1) adhere to fundamental radiological safety requirements (such as effective communication and understanding of work scope, knowledge of actual radiological conditions and potential safety consequence, and conduct of appropriate radiological surveys or evaluations); (2) recognize the potential health and safety consequence of the emergent situation and respond appropriately; and (3) recognize and effectively communicate to management, a situation which delayed defueling activities and resulted in maintaining the reactor in a heightened shutdown risk condition for an extended period. Further, we are concerned that your staff failed to recognize that a substantial potential existed for personnel exposure to airborne radioactivity containing alpha emitters and consequently failed to initiate timely and appropriate personnel exposure evaluation.

Despite, the five "apparent" violations, as in the previously noted events, the NRC has yet to issue any notices of violation.

The incidents cited above are merely exemplary of the type of conduct in which NU has engaged for the past 20 years at all three Millstone units and at Connecticut Yankee. The examples are not meant to be exhaustive of the evidence in support of the contention that NU has knowingly, willingly and recklessly operated Millstone Unit 1, Unit 2, Unit 3 at Waterford, CT, and its Connecticut Yankee Nuclear Power Plant at Haddam Neck, CT, in violation of their respective operating licenses, the regulations of the NRC, and their respective UFSAR's for a prolonged period of time which unnecessarily but significantly compromised public health and safety by eroding the required defense in depth philosophy. In addition to the items specifically cited above, Petitioner requests

that the Commission take administrative notice of the events reflected in the documents listed in the Section entitled, "Evidence in Support of Petition," below and to additional evidence that Petitioner plans to present at the requested public hearings.

Contention No. 2

The evidence is equally overwhelming that NU has knowingly, willingly and intentionally harassed, intimidated and discriminated against its employees who raise safety concerns in violation of United States statutes and NRC regulations for a prolonged period of time. The NRC Office of Investigations has completed one investigation into Petitioner's allegations that he was subjected to HI&D as a result of raising safety concerns about CU-29.[18] It currently is investigating HI&D allegations related to LPCI heat exchanger scale.

In addition to the incidents of HI&D to which Petitioner was subject, and which are discussed in more detail below, the NRC itself has acknowledged, albeit belatedly, that NU has a long history of HI&D. In September 1996, the Millstone Independent Review Group ("MIRG") issued its report entitled, "HANDLING OF EMPLOYEE CONCERNS AND ALLEGATIONS AT MILLSTONE NUCLEAR POWER STATION UNITS 1, 2 & 3 FROM 1985 - PRESENT." The report concluded:

> The MIRG determined that in general, an unhealthy work environment, which did not tolerate dissenting views, and did not welcome or promote a questioning attitude, has existed at Millstone for at least several years. This poor environment has resulted in repeated instances of discrimination and ineffective handling of employee concerns. The vast majority of employee concerns and allegations that were submitted at Millstone represented little safety significance;[19] however many involved potentially important procedural, tagging, or quality

---

[18]/ In response to a request under the Freedom of Information Act for all transcripts of Petitioner's interviews with OI, Petitioner was informed that "[t]hese records have been referred to the Department of Justice for consideration of prosecutive merit." Response to Freedom of Information Act (FOIA) Request No. 96-382, dated October 7, 1996. It is Petitioner's understanding that OI investigative results are not sent to the Department of Justice unless OI has entered a finding of intentional misconduct.

[19]/ Petitioner disputes that the majority of allegations were of "little safety significance." Even assuming it is true that most allegations, taken in isolation, were not safety significant, the cumulation obviously was of great safety significance as evidenced by the fact that none of the plants currently will be allowed to restart without a vote of the Commission.

> assurance (QA) problems, and few were ultimately determined to have safety significance. The unhealthy work environment combined with the significance of substantiated allegations contributed to Millstone being placed on the NRC's watch list in January 1996.

Id. at p. 1.[20] As to the causes of this "unhealthy environment," the MIRG concluded:

> [T]hat these root causes underscored a common theme of top management failure to provide the dynamic and visible leadership needed to bring about required, basic attitude changes. None of the findings of this team are new. Every problem identified during this review had been previously identified to NU management, often by its own self-assessments, yet the same problems continue. This single failure is viewed as being at the core of Millstone's continuing employee concerns.

Id. at p. 2. This root cause, standing alone, demonstrates that NU, as a corporate entity, deliberately and intentionally created the hostile work environment and then allowed it to persist.

Petitioner has been the subject of long-standing and pervasive HI&D. In 1992, Petitioner was the immediate supervisor of Mr. Galatis when he first started pursuing internally NU's full core offloading practice at Millstone Unit 1. Petitioner was essentially the first person in the chain of management to agree that Mr. Galatis was correct in his assertion that Millstone Unit 1 was in violation of its license.

At the same time that he was supporting Mr. Galatis, Petitioner also was pursuing, internally with NU, his concerns about ISAP and CU-29. In December 1993, Petitioner was demoted from his position of Engineering Supervisor after supporting Mr. Galatis and completing a formal evaluation of the CU-29 issue in REF 92-84, which NU management repeatedly rejected but Petitioner ultimately reversed.

---

[20]/ It should be noted that the harassment of Mr. Galatis and Petitioner specifically were excluded from the MIRG study, as was that of several other individuals, including 104 employees who were subject to layoff in January 1996. The NRC, in conjunction with the Department of Labor, conducted an inquiry into the January 1996 layoffs, which is yet to be made public. However, the Department of Labor has concluded that all of the 104 severance agreements contained a clause which was void as against public policy because it required the employees to waive the rights to file a complaint under the Energy Reorganization Act or forfeit their severance benefits.

After his demotion, Petitioner experienced a decline in his performance ratings as he continued to pursue safety-related issues both internally and with the NRC. In the summer of 1995, Petitioner was not selected for a technical position with Public Service Co. of New Hampshire ("PSNH") which is affiliated with NU's Seabrook Nuclear Power Station.

In November 1995, Petitioner was given a psychological evaluation as part of an application for another supervisory position at NU. In December 1995, Petitioner was given his results. Petitioner was advised that his evaluation was unacceptable in eight of 11 categories. Typical of HI&D, Petitioner was advised of incredibly poor interpersonal skills, inferring he was nothing but a liability to NU. NU withheld the written report of the evaluation which Petitioner was promised.

From at least 1993 on, Petitioner has been continually subjected to a hostile environment which was deliberately and intentionally created by NU management in order to discourage employees from identifying and pursuing safety issues. Finally, in August 1996, Petitioner transferred from his engineering position to a senior analyst position in the Employees Concerns Program ("ECP") in hope of extracting himself from the hostile environment. Petitioner continues to be subjected to a hostile environment in his present position by limited participation opportunities for his qualifications and experience.

In further support of this contention, Petitioner requests that the NRC take administrative notice of all documents, transcripts, reports and findings from OI investigations and Department of Labor proceedings involving Eliot Abolofia, Donald Del Core, Sr., Timothy O'Sullivan, Paul Blanch, George Galatis, George Betancourt and Harry Scully.

<u>Contention No. 3</u>

Based on all of the above, as well as the evidence incorporated by reference, *infra*, it is clear that NU cannot provide reasonable assurances that it can cease and desist from engaging in similar activities in the future. To borrow a commonly used cliche, "the promises [of NU] are not worth the paper they are written on." And, the protection of the public health and safety demands more than paper promises. The protection of the public health and safety demands that a clear set of operating conditions be placed on Millstone Units 1, 2 and 3 and Connecticut Yankee with clear penalties to be extracted in the event that those conditions are not met.

Petitioner is by no means the only one to conclude that NU is not capable of properly operating its nuclear power plants. In an audit conducted for the Connecticut Department of Public Utility Control ("DPUC") by the Barrington-Wellesley Group, Inc., the

auditors concluded:

> Neither NU's senior executives nor its Board of Trustees (Board) have exhibited the leadership and vision necessary to address the fundamental needs of a well-performing nuclear program. NU and its Board had sufficient information to realize that decisive action was necessary to address the deteriorating performance in NU's nuclear operation, especially at MP1 and MP2. . . .
>
> NU does not have a coherent, long-term strategy for its nuclear operations. . . .

See, "A Focused Management Audit of the Connecticut Light and Power Company's Nuclear Operations," August 30, 1996, Executive Summary at p. 2.

Similarly, another DPUC audit conducted by R.C. Brown & Associates, Inc., concluded:

> RCB&A's review of the historical record, as well as the results of interviews with NU personnel, indicated that the issues cited in designating the Millstone site to its Watch List have been matters of discussion between NU and the NRC for a number of years, in some cases dating back to the late 1980s. To address such NRC concerns, NU developed a number of improvement initiatives over the years. However, such initiatives suffered from a lack of management direction and commitment, with the result that such initiatives were generally ineffective in addressing and resolving the NRC's concerns.
>
> Between 1992 and 1994, NU undertook a Performance Enhancement Program ("PEP"), consisting of 42 "Action Plans" designed to address weaknesses in management practices; programs and processes; and performance assessment. However, the PEP suffered from lack of direction and support form (sic) senior management, and was terminated without effective validation that the program had achieved its intended results.
>
> In late 1993, following a significant safety-related incident at Millstone Unit 2, NU instituted a significant reorganization at the Millstone site, including the appointment of a new Senior Vice-President, Millstone Station. The new Senior V.P. established his own corrective action initiative, designated the Improving Station Performance ("ISP") program. The ISP featured initiatives in the areas of effective employee communications; leadership development; effective corrective action; procedure development and use; work planning and control; and operations. As was the case

with the PEP, the ISP suffered from a lack of support from senior management, failed to meet its most important objectives and, as of early 1996, had been largely abandoned.

See, "Focused Audit of the Connecticut Light and Power Company: Nuclear Operations," R.C. Brown & Associates, Inc., December 31, 1996, at pp v-13 to v-14.

At a meeting of the Connecticut Nuclear Energy Advisory Committee ("NEAC") on February 20, 1997, an NU official stated that Millstone would try to restart three units within one year—a feat never accomplished by anyone in the world. Of course, NU is quite familiar with achievements not previously accomplished. It is indeed an elite group that, in the course of one year, is denied permission to restart four units. But, one can only question the absolute folly of even considering the prospect that NU would be allowed to restart all three units in a one year period of time. Yet, that is what the NRC is contemplating even at this moment, while it allows the previous misconduct of NU to go unpunished.

Nor is the public's confidence in the ability of NU to reform its ways in the future restored by the fact that in December 1996 six of seven NU employees who applied for a license to operate Millstone Unit 1 failed the examination. The Initial Operator Licensing Examination Report, dated February 7, 1997 concluded:

> The applicants were poorly prepared for the examination, and as a result, 6 of 7 applicants failed the examination. One SRO instant applicant passed by the examination by a very small margin. Significant generic weaknesses were noted in both the written and operating test, and four of the seven applicants had written scores of 72 or below. Significant weaknesses were identified during the operating test related to operating reactivity controls, diagnosis of instrument failures, and diagnosis of ECCS injection status.

Equally, NU is poorly prepared for the "test" ahead of it. But this is not a "test."

Plans on paper, reorganizations, and new management initiatives have been tried in the past. All have had the same result. They merely have provided NU with a guise to go about business in its usual manner elevating cost over safety. There is absolutely no reason to believe that this time will be any different. The only thing that can make this time different is to clearly spell out the rules of the game, and impose stiff and swift penalties in the event the rules are violated. This is the only manner in which the NRC can give the public some reasonable assurance that NU will not repeat its past behavior.

Evidence in Support of Petition

In addition to the evidence specifically cited above, a large body of factual evidence, on which this petition is based, is currently in existence and in the possession of the NRC. Incorporated, by reference, into the factual basis for this petition are the following documents:

1. Petition submitted pursuant to 10 C.F.R. § 2.206 on behalf of George J. Galatis and We the People, Inc., of the United States, dated August 21, 1995;

2. Supplemental petition submitted pursuant to 10 C.F.R. § 2.206 on behalf of George J. Galatis and We the People, Inc., of the United States, dated August 28, 1995;

3. Report of U.S. Nuclear Regulatory Commission, Office of Inspector General, "NRC FAILURE TO ADEQUATELY REGULATE - MILLSTONE UNIT 1," Case No. 95-77I, dated December 21, 1995;

4. Report of U.S. Nuclear Regulatory Commission, Office of Inspector General, "NRC STAFF ACTIONS TO ADDRESS NORTHEAST UTILITIES SYSTEM (NU) 1991 SELF-ASSESSMENTS," Case No. 96-02S, dated May 31, 1996;

5. Report of U.S. Nuclear Regulatory Commission, Office of Inspector General, "NRC HANDLING OF ISSUES RELATED TO REFUELING OPERATIONS AT MILLSTONE UNIT 1," Case No. 96-05S, dated July 23, 1996;

6. Report of U.S. Nuclear Regulatory Commission, Office of Inspector General, "NRC STAFF ACTIONS TO ADDRESS CU-29 ISOLATION VALVE ISSUE," Case No. 96-06S, dated September 3, 1996;

7. Report(s) of U.S. Nuclear Regulatory Commission, Office of Investigations, regarding allegations of intentional wrongdoing by George J. Galatis, date(s) uncertain;

8. Report(s) of U.S. Nuclear Regulatory Commission, Office of Investigations, regarding allegations of intentional wrongdoing by Albert A. Cizek, date(s) uncertain;

9. Report(s) of U.S. Nuclear Regulatory Commission, Office of Investigations, regarding allegations of intentional wrongdoing by George Betancourt, date(s) uncertain;

10. Millstone Unit No. 1, Safety Systems Functional Inspection of the Condensate/Feedwater\Feedwater Coolant Injection System, December 16, 1988;

11. NEO Performance Task Group Report, NU, September 1991;

12. Final Report of the <u>Procedure Compliance Task Force</u> at Millstone Point Station, date uncertain;

13. Evaluation of Millstone Nuclear Power Station, Institute of Nuclear Power Operations, December 1, 1992;

14. Report of Self-Assessment Task Force, NU, May 1994;

15. Millstone Independent Review Group, Handling of Employee Concerns and Allegations at Millstone Nuclear Power Station, Units 1, 2 & 3 from 1985 - Present, September 1996.

Conclusion

For the reasons cited fully above, Petitioner respectfully requests that the NRC institute a proceeding pursuant to 10 C.F.R. § 2.202 to modify the operating licenses of Millstone Units 1, 2 and 3 and Connecticut Yankee as described above.

Respectfully submitted,

Ernest C. Hadley

cc: A Cizek
H. Bell, IG
B. Kenyon, Pres., NU