October 16, 2020

Margaret Doane
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington D.C. 20555-0001

10 C.F.R. § 2.206 REQUEST TO SUSPEND THE OPERATING LICENSES OF BOILING WATER REACTOR MARK I UNITS UNTIL THE HARDENED CONTAINMENT VENTS PRESENTLY INSTALLED IN SUCH UNITS ARE REPLACED WITH VENTS CAPABLE OF DISCHARGING THE TOTAL AMOUNT OF THERMAL ENERGY (HEAT) THAT WOULD BE GENERATED OVER SOME PERIODS OF TIME DURING A SEVERE ACCIDENT

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October 16, 2020

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the Matter of:	:	TO: MARGARET DOANE Executive Director for Operations U.S. Nuclear Regulatory Commission Washington D.C. 20555-0001
TENNESSEE VALLEY AUTHORITY (Browns Ferry Nuclear Plant Units 1, 2, and 3; Docket Nos. 50-259, 50-260, and 50-296)	: : : :	Docket No. ______
DUKE ENERGY PROGRESS, LLC (Brunswick Steam Electric Plant, Units 1 and 2; Docket Nos. 50-325 and 50-324)		
NEBRASKA PUBLIC POWER DISTRICT (Cooper Nuclear Station; Docket No. 50-298)		
EXELON GENERATION COMPANY, LLC (Dresden Nuclear Power Station, Units 2 and 3; Docket Nos. 50-237 and 50-249)		
SOUTHERN NUCLEAR OPERATING COMPANY, INC. (Edwin I. Hatch Nuclear Plant, Units 1 and 2; Docket Nos. 50-321 and 50-366)		
DTE ELECTRIC COMPANY (Enrico Fermi Nuclear Generating Station; Docket No. 50-341)		
PSEG NUCLEAR, LLC (Hope Creek Generating Station; Docket No. 50-354)		
EXELON GENERATION COMPANY, LLC (James A. FitzPatrick Nuclear Power Plant; Docket No. 50-333)		
NORTHERN STATES POWER COMPANY (Monticello Nuclear Generating Plant; Docket No. 50-263)		
NINE MILE POINT NUCLEAR STATION, LLC (Nine Mile Point Nuclear Station, Unit 1; Docket No. 50-220)		
EXELON GENERATION COMPANY, LLC (Peach Bottom Atomic Power Station, Units 2 and 3; Docket Nos. 50-277 and 50-278)		

EXELON GENERATION COMPANY, LLC
(Quad Cities Nuclear Power Station, Units 1 and 2; Docket Nos. 50-254 and 50-265)

MARK LEYSE and BEYOND NUCLEAR
Petitioners

10 C.F.R. § 2.206 REQUEST TO SUSPEND THE OPERATING LICENSES OF BOILING WATER REACTOR MARK I UNITS UNTIL THE HARDENED CONTAINMENT VENTS PRESENTLY INSTALLED IN SUCH UNITS ARE REPLACED WITH VENTS CAPABLE OF DISCHARGING THE TOTAL AMOUNT OF THERMAL ENERGY (HEAT) THAT WOULD BE GENERATED OVER SOME PERIODS OF TIME DURING A SEVERE ACCIDENT

I. REQUEST FOR ACTION

This petition for an enforcement action is submitted pursuant to 10 C.F.R. § 2.206 by Mark Leyse[1] and Beyond Nuclear. 10 C.F.R. § 2.206(a) states that "[a]ny person may file a request to institute a proceeding pursuant to § 2.202 to modify, suspend, or revoke a license, or for any other action as may be proper."

Petitioners first request that the United States Nuclear Regulatory Commission ("NRC") suspend the operating licensees of boiling water reactor ("BWR") Mark I units until the hardened containment vents presently installed in such units are replaced with vents that are capable of discharging the total amount of thermal energy (heat) that would be generated over some periods of time during a severe accident. Secondly, Petitioners request that the NRC revoke the licenses of any BWR Mark I units that are not modified to fulfill Petitioners' first request.

[1] Mark Leyse is the author of this 10 C.F.R. § 2.206 petition for an enforcement action, excepting Sections V and VI, which he coauthored. Sections V and VI were primarily written by David Lochbaum. The author thanks David Lochbaum, M. V. Ramana, Frank N. von Hippel, and Robert H. Leyse for their helpful comments and suggestions. The author also thanks Paul Gunter of Beyond Nuclear for his help and for shining a light on the safety issues the petition raises. Leyse takes full responsibility for the content of this petition: he is solely responsible for any errors.

II. STATEMENT OF PETITIONERS' INTEREST

On March 15, 2007, Mark Leyse submitted a 10 C.F.R. § 2.802 petition for rulemaking, PRM-50-84,[2] to the NRC. PRM-50-84 requested: 1) that the NRC make new regulations to help ensure licensees' compliance with 10 C.F.R. § 50.46(b) emergency core cooling systems ("ECCS") acceptance criteria and 2) to amend Appendix K to Part 50, "ECCS Evaluation Models," I(A)(1), "The Initial Stored Energy in the Fuel."

In 2008, the NRC decided to consider the issues raised in PRM-50-84 in its rulemaking process.[3] And in 2009, the NRC published "Performance-Based Emergency Core Cooling System Acceptance Criteria," which gave advanced notice of a proposed rulemaking, addressing four objectives: the fourth being the issues raised in PRM-50-84.[4] In 2012, the NRC Commissioners voted unanimously to approve a proposed rulemaking—revisions to Section 50.46(b), which will become Section 50.46(c)—that was partly based on the safety issues Leyse raised in PRM-50-84.[5]

Beyond Nuclear is a nonprofit 501(c)(3) membership organization. Beyond Nuclear is organized exclusively for charitable, scientific, and educational purposes. Specifically, the organization aims to educate and activate the public about the connections between nuclear power and nuclear weapons and the need to abandon both to safeguard our future. Beyond Nuclear advocates for an energy future that is sustainable, benign, and democratic. The organization works with diverse partners and allies to provide its members, the public, government officials, and the media with the critical information necessary to move humanity toward a world beyond nuclear.

III. FACTS CONSTITUTING THE BASIS FOR PETITIONERS' REQUEST

III.A. Design Flaws of BWR Mark I Primary Containments

In September 1972, Stephen Hanauer, a well-known nuclear safety analyst working for the U.S. Atomic Energy Commission ("AEC")—the NRC's predecessor agency—sent a

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[2] Mark Leyse, PRM-50-84, March 15, 2007, (ADAMS Accession No. ML070871368).

[3] Federal Register, Vol. 73, No. 228, "Mark Edward Leyse; Consideration of Petition in Rulemaking Process," November 25, 2008, pp. 71564-71569.

[4] Federal Register, Vol. 74, No. 155, "Performance-Based Emergency Core Cooling System Acceptance Criteria," August 13, 2009, pp. 40765-40776.

[5] NRC, Commission Voting Record, Decision Item: SECY-12-0034, Proposed Rulemaking—10 CFR 50.46(c): Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (RIN 3150-AH42), January 7, 2013, (ADAMS Accession No. ML13008A368).

memo to the AEC's director of licensing, John F. O'Leary, and others recommending that the AEC stop licensing the pressure-suppression containments used in General Electric's BWR Mark I design, warning that they are prone to overpressurization in the event of a severe accident.[6] The volume of a Mark I containment is too small to safely accommodate a reactor as it melts down, generating thousands of megajoules of thermal energy, thousands of kilograms of steam, and hundreds of kilograms of noncondensable hydrogen gas.[7]

Hanauer wanted to have pressure-suppression containments banned at a future date, not to revoke the licenses of the dozens of reactors with such a design that were already in operation or under construction. Following up on Hanauer's memo, Joseph M. Hendrie, a senior official in the AEC's licensing division, wrote O'Leary, saying he saw merit in the "idea to ban pressure-suppression containment schemes." However, Hendrie also said publicly admitting the design is flawed "would throw into question the continued operation of licensed plants, would make unlicensable the GE and Westinghouse ice condenser plants [with pressure-suppression containments] now in review, and would generally create more turmoil than I can stand thinking about." At a time when the AEC's competence was in question, Hendrie feared disclosing the defects of pressure-suppression containments might spell "the end of nuclear power."[8]

The AEC ended up concealing information about the inherent flaws of pressure-suppression containments. The agency feared disclosing their defects would lead to lawsuits demanding the shutdown and termination of construction of reactors with the flawed design.[9]

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[6] Stephen Hanauer, AEC nuclear safety analyst, memorandum regarding "Pressure-Suppression Containments" to senior AEC staff members, John F. O'Leary, Director of Licensing, *et al*., September 20, 1972. Daniel Ford, *The Cult of the Atom: The Secret Papers of the Atomic Energy Commission*, (New York: Simon and Schuster, 1982), pp. 193-194.

[7] Hydrogen gas is noncondensable in a nuclear plant's temperature range. Hydrogen gas will liquefy if it is chilled to minus-253 degrees Celsius (minus-423 degrees Fahrenheit), at sea level.

[8] Joseph M. Hendrie, senior AEC staff member, memorandum regarding Stephen Hanauer's assertion that pressure-suppression containments have safety disadvantages to John F. O'Leary, Director of Licensing, September 25, 1972. Daniel Ford, *The Cult of the Atom: The Secret Papers of the Atomic Energy Commission*, (New York: Simon and Schuster, 1982), pp. 194-195.

[9] Daniel Ford, *The Cult of the Atom: The Secret Papers of the Atomic Energy Commission*, (New York: Simon and Schuster, 1982), pp. 194-195.

In addition to design flaws of BWR Mark I primary containments, the design of Mark I units as a whole is flawed because the spent fuel pool is located within the reactor building, in proximity to the primary containment, making it vulnerable to hydrogen explosions. (The base of the 12-meter-deep pool is elevated approximately 15 meters (50 feet) above ground level; the top of the pool is located at the level of the operating floor, elevated approximately 27 meters (90 feet) above ground level.[10]) At Fukushima Daiichi, the force of hydrogen explosions flung tons of rubble and plant equipment into three of the plant's spent fuel pools. At Unit 3, a gigantic fuel handling machine landed in the spent fuel pool.[11]

In response to the hydrogen explosion that occurred in the Three Mile Island accident, the NRC—which had supplanted the AEC in January 1975[12]—issued a requirement in 1981 that BWR Mark I (and Mark II) units operate with a low concentration of oxygen in their primary containments. The agency requires a concentration low enough to prevent hydrogen explosions (less than four percent oxygen by volume).[13] Nitrogen gas is used to displace oxygen from the containments because nitrogen is inexpensive and nontoxic. Nitrogen gas would certainly be helpful for preventing hydrogen explosions within the primary containment during a severe accident; however, nitrogen is a noncondensable gas in the temperature range of a nuclear plant.[14] The presence of a large quantity of nitrogen within the primary containment would

[10] Allan S. Benjamin *et al.*, "Spent Fuel Heatup Following Loss of Water During Shortage," Sandia Laboratories, NUREG/CR-0649, March 1979, (ADAMS Accession No. ML120960637), p. 15.

[11] Committee on Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Plants *et al.*, *Lessons Learned from the Fukushima Accident for Improving Safety and Security of U.S. Nuclear Plants: Phase 2*, (Washington DC: The National Academy Press, 2016), pp. 33, 54.

[12] Daniel Ford, *The Cult of the Atom: The Secret Papers of the Atomic Energy Commission*, (New York: Simon and Schuster, 1982), p. 167.

[13] The NRC requires that BWR Mark I (and Mark II) units operate with an "inerted" containment atmosphere. See NRC Policy Statement, "Combustible Gas Control in Containment," Federal Register, Vol. 68, No. 179, September 16, 2003, pp. 54123, 54141.

[14] Nitrogen gas will liquefy if it is chilled to minus-196 degrees Celsius (minus-320 degrees Fahrenheit), at sea level.

exacerbate the problems of containment overpressurization that are likely to occur in the event of a severe accident.[15]

In September 1989, the NRC publicly acknowledged that a BWR Mark I pressure-suppression containment might not withstand the high pressures and extreme temperatures of a severe accident. However, at the time, the NRC merely issued non-legally binding guidance, Generic Letter 89-16, recommending that owners of BWR Mark I units "on their own initiative" install a hardened vent to the wetwell of the primary containment for the purposes of depressurizing and discharging heat in the event of an accident.[16] The NRC reasoned that a hardened wetwell vent would enhance the ability of plant workers to either prevent or mitigate a severe accident.[17]

III.B. Post-Fukushima Requirements for BWR Mark I Primary Containments

In the Fukushima Daiichi accident, hardened vents installed in BWR Mark I primary containments failed to prevent hydrogen gas from leaking into reactor buildings and exploding. The Fukushima accident revealed potential deficiencies of the hardened vents that had been voluntarily installed in U.S. BWR Mark I units. The NRC was concerned that the U.S. vents did not have a dependable design. On March 12, 2012, the NRC issued Order EA-12-050, requiring that new "reliable" hardened vents be installed in the primary containments of all BWR Mark I (and Mark II) units to assist strategies for preventing severe accidents.[18]

The following year, on June 6, 2013, the NRC issued Order EA-13-109, stipulating additional requirements for the new hardened vents: mainly, that they be

[15] T. Okkonen, OECD Nuclear Energy Agency, "Non-Condensable Gases in Boiling Water Reactors," NEA/CSNI/R(94)7, May 1993.

[16] The hardened wetwell vents that were installed in the United States before the Fukushima Daiichi accident did not have standardized features, were not subject to inspection by the NRC for proper maintenance and assured operability, and did not have an independent train of backup power sources to help ensure remote operation in the event of a station blackout (that is, a scenario in which a nuclear power plant loses both grid-connected and onsite backup alternating current power).

[17] NRC, "Installation of a Hardened Wetwell Vent," Generic Letter 89-16, September 1, 1989, (ADAMS Accession No. ML031140220), p. 1.

[18] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," EA-12-050, March 12, 2012, (ADAMS Accession No. ML12054A694).

capable of operating under severe accident conditions.[19] The NRC specified that the vents must have the capacity to handle "the elevated temperatures, pressures, radiation levels, and combustible gas concentrations, such as hydrogen and carbon monoxide, associated with accidents involving extensive core damage, including accidents involving a breach of the reactor vessel by molten core debris."[20] (Molten core materials would enter the primary containment's drywell if they melted through the reactor pressure vessel.)

Order EA-13-109 required nuclear plant owners to install two new hardened vents in the primary containments of each of the nuclear fleet's BWR Mark I (and Mark II) units—one vent in the wetwell, a second in the drywell.[21] It is best to vent from the wetwell: the suppression pool water can scrub and retain a portion of the reactor's radioactive fission products, with the exception of noble gases,[22] partly decontaminating releases to the outside environment.[23] Plant workers are directed to vent from the drywell, which releases radionuclides directly into the environment, only if the wetwell vent becomes inoperable.

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[19] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321).

[20] *Id*., pp. 3-4.

[21] Order EA-13-109 offers plant owners an alternative to installing a hardened containment vent in the drywell, providing they "develop and implement a reliable containment venting strategy that makes it unlikely that [plant workers] would need to vent from the containment drywell before alternate reliable containment heat removal and pressure control is reestablished." See NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321), Attachment 2: "Requirements for Reliable Hardened Vent Systems Capable of Operation under Severe Accident Conditions at Boiling-Water Reactor Facilities with Mark I and Mark II Containments," p. 4.

[22] Noble gases (helium, neon, argon, krypton, xenon, radon) have only a slight tendency to be chemically reactive.

[23] The effectiveness of suppression pool scrubbing depends on several factors, including the accident scenario and the state of the suppression pool itself. Decontamination may range from none to greater than 99.9 percent effective. According to conservative analyses, decontamination for Mark I designs is typically at least 80 percent effective. This value pertains to fission products other than noble gases, which suppression pool water does not retain. See R. Jack Dallman *et al*., "Filtered Venting Considerations in the United States," May 17-18, 1988, CSNI Specialists Meeting on Filtered Vented Containment Systems, Paris France, p. 4.

III.C. The Vents the NRC Requires Would Not Prevent a BWR Mark I Primary Containment from Failing in a Severe Accident

Order EA-13-109 stipulated that the new BWR Mark I (and Mark II) vents must have the capacity to handle a continuous thermal energy (heat) input at a rate equal to one percent of the reactor's rated maximum thermal power, while maintaining the primary containment at a pressure lower than its design pressure and pressure limit.[24]

There is good reason to believe that the hardened vents required by Order EA-13-109 would fail to protect a BWR Mark I primary containment from being compromised in a severe accident. A report prepared for the NRC by Idaho National Engineering Laboratory ("INEL") in the late 1980s states that in a severe accident the thermal energy generated by the chemical reaction of steam and metals within the reactor, primarily zirconium, may be "several times" as great as that generated by decay heating. INEL maintained that a vent intended to remove heat from and depressurize a BWR Mark I primary containment under severe accident conditions should be able to handle a continuous thermal energy input at a rate equal to *seven percent* of the reactor's rated maximum thermal power.[25]

Idaho National Engineering Laboratory Versus the NRC on the Vent Capacities Needed for Handling the Thermal Energy Generated from Decay Heating under Non-Severe Accident Conditions

Order EA-12-050 (which was cancelled by Order EA-13-109) required hardened containment vents for BWR Mark I (and Mark II) units to have the capability to assist

[24] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321), Attachment 2: "Requirements for Reliable Hardened Vent Systems Capable of Operation under Severe Accident Conditions at Boiling-Water Reactor Facilities with Mark I and Mark II Containments," p. 2.

[25] This value is based on calculations performed for the two BWR Mark I units at Peach Bottom Atomic Power Station, which at the time were both rated at 3,293 megawatts thermal (MWt). See K. C. Wagner *et al*., "An Overview of BWR Mark-1 Containment Venting Risk Implications," Idaho National Engineering Laboratory, NUREG/CR-5225, June 1989, (ADAMS Accession No. ML101870670), p. 25. See also NRC, "Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 2," October 18, 1994, (ADAMS Accession No. ML011490143), p. 1. Additionally, see NRC, "Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 3," July 18, 1995, (ADAMS Accession No. ML021580312), p. 1.

strategies for preventing severe accidents. Order EA-12-050 explained that "the NRC has…determined that Licensees should promptly begin the implementation of short-term actions relating to reliable hardened vents and to focus these actions on improvements that will *assist in the prevention of core damage*. As such, this Order requires Licensees to take the necessary actions to install reliable hardened venting systems in BWR facilities with Mark I and Mark II containments *to assist strategies relating to the prevention of core damage*" [emphasis added].[26]

Order EA-12-050 required new hardened containment vents to have the capacity "to vent the steam-energy equivalent of 1 [one] percent of licensed-rated thermal power (unless a lower value is justified by analyses), and be able to maintain containment pressure below the primary containment design pressure."[27] (Keep in mind that this requirement was *not* for handling the thermal energy generated during a severe accident.) The NRC has explained that after the reactor shuts down, the wetwell's suppression pool is typically capable of absorbing the nuclear fuel's decay heat for a period of three hours. After three hours, the thermal power from continuous decay heat generation has typically decreased to less than one percent of the reactor's rated maximum thermal power.[28]

Idaho National Engineering Laboratory ("INEL") pretty much agrees with the NRC on what the capacity of a hardened containment vent needs to be in order to handle the thermal energy that would be generated in an event that does not transform into a severe accident. INEL states: "During sequences where the reactor is scrammed, such as the TW [loss of long-term decay heat removal] or station blackouts, *the decay heat energy in the core is less than 1.5% of the rated thermal core power at 1 h [one hour]*. Therefore, an 8- to 10-in. vent line…should be able to remove the decay heat energy

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[26] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," EA-12-050, March 12, 2012, (ADAMS Accession No. ML12054A694), p. 5.

[27] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," EA-12-050, March 12, 2012, (ADAMS Accession No. ML12054A694), Attachment 2: "Requirements for Reliable Hardened Vent Systems at Boiling-Water Reactor Facilities with Mark I and Mark II Containments," p. 1.

[28] NRC, "Compliance with Order EA-12-050, Reliable Hardened Containment Vents: Interim Staff Guidance," Attachment I, August 29, 2012, (ADAMS Accession No. ML12229A475), p. 5. Note that the capacity requirement for a hardened containment vent stipulated in Order EA-13-109 is the same as that stipulated in Order EA-12-050.

from a plant with the same thermal power rating as Peach Bottom" [emphasis added].[29] (At that time the two BWR Mark I units at Peach Bottom Atomic Power Station were both rated at 3,293 megawatts thermal.)

Order EA-12-050, with its requirement that a hardened containment vent be capable of handling one percent of a reactor's licensed-rated thermal power under non-severe accident conditions, was actually based on sound science.

Idaho National Engineering Laboratory Versus the NRC on the Vent Capacities Needed for Handling the Total Amount of Thermal Energy Generated under Severe Accident Conditions

Order EA-13-109 requires hardened containment vents for BWR Mark I (and Mark II) units to have the capability to operate under severe accident conditions. Order EA-13-109 specifies that the new vents must have the capacity to handle "the elevated temperatures, pressures, radiation levels, and combustible gas concentrations, such as hydrogen and carbon monoxide, *associated with accidents involving extensive core damage, including accidents involving a breach of the reactor vessel by molten core debris*" [emphasis added].[30]

Order EA-13-109 required new hardened containment vents to have the capacity "to vent the steam-energy equivalent of one (1) percent of licensed-rated thermal power (unless a lower value is justified by analyses), and be able to restore and then maintain containment pressure below the primary containment design pressure and the primary containment pressure limit."[31] These requirements for the capacity of hardened containment vents are essentially the same as those required by Order EA-12-050, which is problematic because requirements of Order EA-13-109 do not account for the total

[29] K. C. Wagner *et al.*, "An Overview of BWR Mark-1 Containment Venting Risk Implications," Idaho National Engineering Laboratory, NUREG/CR-5225, June 1989, (ADAMS Accession No. ML101870670), p. 25.

[30] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321), pp. 3-4.

[31] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321), Attachment 2: "Requirements for Reliable Hardened Vent Systems Capable of Operation under Severe Accident Conditions at Boiling-Water Reactor Facilities with Mark I and Mark II Containments," p. 2.

amount of thermal energy that would be generated by the chemical reaction of steam and metals within the reactor, primarily zirconium, during a severe accident.

INEL explained: "If core degradation or core-concrete interactions have commenced, the vent system must be able to remove decay heat energy *plus the chemical energy from metal-water reactions*. The energy from metal-water reaction can represent several times the amount of decay energy. However, it is expected that an 18-in. line, which can remove approximately 7% of the rated thermal core power, could adequately depressurize the system. This was confirmed by calculations performed for the Peach Bottom venting study" [emphasis added].[32] (The study referred to is: D. J. Hanson *et al*., "Containment Venting Analysis for the Peach Bottom Nuclear Power Plant," NUREG/CR-4696, EGG-2464, December 1986.)

Order EA-13-109, with its requirement that a hardened containment vent be capable of handling one percent of a reactor's licensed-rated thermal power under severe accident conditions, is not based on sound science.

One might suspect the NRC intentionally shirked its duty to protect public safety in the interest of placating industry by ensuring Order EA-13-109, with its grossly inadequate requirements, spared plant owners the expense of providing an actual solution to the inherent flaws of BWR Mark I containments. Why is it that the esteemed agency neglected to consider information it itself published in its very own NUREG series in a report—"An Overview of BWR Mark-1 Containment Venting Risk Implications"—it commissioned from Idaho National Engineering Laboratory in the late 1980s, indicating post-Fukushima vents need a significantly higher capacity than Order EA-13-109 requires?

On its website the NRC claims that the new "reliable" vents required by Order EA-13-109 are assured to "function in the conditions following reactor core damage." The agency also informs the public that the new vents have been installed at all BWR Mark I (and II) units "and verified by NRC inspectors."[33] The NRC's assurances are

[32] K. C. Wagner *et al*., "An Overview of BWR Mark-1 Containment Venting Risk Implications," Idaho National Engineering Laboratory, NUREG/CR-5225, June 1989, (ADAMS Accession No. ML101870670), p. 25.
[33] NRC, "Containment Venting System: Improved Fukushima-Style Designs," (available at: https://www.nrc.gov/reactors/operating/ops-experience/post-fukushima-safety-

based on a fiction that in severe accidents vast amounts of thermal energy are NOT generated by the reactor's contents, primarily its zirconium, burning in steam. The agency should promptly issue a correction, informing the public that the new vents cannot be relied upon to save the day in the event of a severe accident, preventing a Fukushima Daiichi-type release of radioactive material on American soil.

The NRC should suspend the operating licensees of BWR Mark I units until the hardened containment vents presently installed in such units are replaced with vents that are capable of discharging the total amount of thermal energy that would be generated under the harshest of severe accident conditions, which is far greater than that solely generated by decay heating. If the NRC balks, perhaps state officials—like those in New York—who decided to subsidize BWR Mark I units in economic peril with tens of millions of dollars each year,[34] should consider canceling the subsidies.

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INEL recommended that hardened vents be capable of removing seven times as much thermal power as the NRC requires. Nonetheless, even the vents INEL proposed would be incapable of handling certain severe accident scenarios. For example, flooding a melting-down reactor with coolant water—a standard severe accident procedure—generates an immense amount of thermal energy for a brief period of time, potentially

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enhancements/containment-venting-system.html : last visited on 09/02/20) These quotes are also available in NRC, "Safety Enhancements after Fukushima," December 2018, (ADAMS Accession No. ML18355A806), p. 6.

[34] New York officials have decided to subsidize two BWR Mark I units—James A. FitzPatrick Nuclear Power Plant and Nine Mile Point Nuclear Station, Unit 1—with billions of dollars over a period of a dozen years. See Patrick McGeehan, "New York State Aiding Nuclear Plants With Millions in Subsidies," *The New York Times*, August 1, 2016. (available at: https://www.nytimes.com/2016/08/02/nyregion/new-york-state-aiding-nuclear-plants-with-millions-in-subsidies.html : last visited on 03/03/20)

New York is subsidizing a total of four reactors, including Nine Mile Point Nuclear Station, Unit 2, a BWR Mark II unit. Although BWR Mark II units are not a subject of this petition, the vents required by Order EA-13-109 cannot be guaranteed to prevent the failure of a BWR Mark II containment in the event of a severe accident. Hence, New York is spending billions of dollars subsidizing a total of three reactors that would likely release large amounts of radioactive material in the event of a severe accident.

generating at rates that are greater than 25 percent of a reactor's rated maximum thermal power.[35]

III.C.1. The NRC's Explanation of the Basis for the Capacity of the Vents Required by Order EA-13-109

On July 25, 2013, Robert H. Leyse, the father of the author of this petition, wrote the NRC asking the agency "to provide a list of references that document the basis for" the agency's decision stipulating that the vents required by Order EA-13-109 have the capacity to vent the steam-energy equivalent of one percent of the reactor's licensed-rated thermal power. In addition to his request, Leyse stated: "Certainly, NRC should realize that the amount of hydrogen produced at Fukushima as well as the timing and rate of hydrogen production was not related to 1 percent of the operating power levels of those units."[36] Leyse was essentially pointing out the fact that large amounts of hydrogen were rapidly generated in the Fukushima Daiichi accident means large amounts of thermal energy were also rapidly generated, sometimes at rates exceeding one percent of each affected reactor's rated maximum thermal power. In other words, the vents required by Order EA-13-109 lack the capacity to handle severe accident conditions like those that occurred during the Fukushima Daiichi accident.

The NRC responded to Leyse's request on August 9, 2013 in a short document titled, "Basis for Venting Capacity in Order EA-13-109, 'Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions'."[37] The NRC reiterates its explanation that the venting capacity stipulated by Order EA-13-109 is based on its belief that a vent needs to have the capacity to remove the quantity of decay heat that is generated three hours after the reactor shuts down in an accident.

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[35] This percentage would be for a BWR rated at a maximum power of 2,700 MWt.

[36] Robert H. Leyse, e-mail sent to the NRC's Office of Public Affairs, July 25, 2013, (ADAMS Accession No. ML13326B061).

[37] NRC, "Basis for Venting Capacity in Order EA-13-109, 'Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions'," August 9, 2013, (ADAMS Accession No. ML13221A011).

The NRC states:

> Further support for selecting a value corresponding to the decay heat generation rate at 3 hours after shutdown is that suppression pool water mass relative to reactor licensed power provides at least 3 hours of decay heat absorption capacity before reaching saturation temperature (at atmospheric pressure). When a suppression pool reaches saturation, some alternate means of removing additional decay heat energy from containment is needed to avoid further containment pressure increase. Since the decay heat level after the pool reached saturation would be less than 1% of licensed thermal power, a vent capable of discharging the equivalent steam flow from a containment at design pressure to the atmosphere is considered adequate for the pressure control (heat removal) needed to preserve the ability to return containment to a leak tight configuration when normal or other reliable closed loop means of decay heat removal subsequently became available.[38]

As discussed in Section III.C of this petition, the requirement that a hardened containment vent be capable of handling one percent of a reactor's licensed-rated thermal power is adequate for operation under *non*-severe accident conditions. However, such a vent does not have the capacity to handle the vast quantities of thermal energy that would be generated under severe accident conditions.

In "Basis for Venting Capacity in Order EA-13-109," the NRC offers a bizarre justification for its decision to stipulate such a limited capacity for vents expected to perform under severe accident conditions.

The NRC states:

> The progression of an accident up to and including significant core damage would likely include the need to vent the containment multiple times to prevent a failure of Mark I or Mark II containments due to overpressure conditions. The initial venting operations would likely be performed prior to core damage and the presence of significant amounts of hydrogen and other noncondensible gases and the venting capacity requirements in EA-12-050 were therefore maintained in the revised order, EA-13-109. *In addition, the sizing and design of the venting path would not change significantly for severe accident conditions since the credible hydrogen production rates would require a few percent of the capacity of a system designed for 1% steam flow at design pressure.* The presence of hydrogen and other noncondensible gases could result in higher containment pressures for some scenarios but would unlikely challenge

[38] *Id*., p. 2.

the structural integrity of the containment given the design margin available [emphasis added].[39]

The NRC acknowledges that hydrogen would be generated by chemical reactions under severe accident conditions, also stating that the presence of hydrogen in the primary containment might increase containment pressures. However, the NRC fails to consider that the same chemical reactions that generate hydrogen under severe accident conditions also generate thermal energy. (Of course, a small amount of hydrogen would also be generated by water radiolysis.[40])

Large quantities of thermal energy may be generated in a BWR severe accident. For example, if 1,000 kilograms (kg) of hydrogen were generated by the zirconium-steam reaction in a BWR severe accident—which may have occurred in the Fukushima Daiichi Unit 3 accident[41]—it would mean that 146,785 megajoules (MJ) of thermal energy were also generated.[42] Converting 146,785 MJ of thermal energy to electricity with the efficiency of a power plant, about 33 percent, results in about 48,930 MJ of electrical energy, which is equal to about 13,590 kilowatt-hours (kWh) of electricity: enough electricity to power more than 450 average American homes for a full day.[43]

The NRC's explanation of its basis for the venting capacity it stipulated in Order EA-13-109 displays ignorance of the zirconium-steam reaction. The NRC should learn

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[39] *Id*.

[40] Water radiolysis may generate from 0.001 to 0.05 kg of hydrogen per second. See E. Bachellerie *et al*., "Generic approach for designing and implementing a passive autocatalytic recombiner PAR-system in nuclear power plant containments," Nuclear Engineering and Design, Vol. 221, Nos. 1-3, April 2003, p. 158.

[41] Sandia National Laboratories conducted analyses of the Fukushima Daiichi Unit 3 accident, estimating that during the accident, steam reactions with reactor core contents—zirconium (cladding and channel boxes), stainless steel, and boron carbide—generated from 970 to 1,350 kilograms of hydrogen. See Jeffrey Cardoni *et al*., "MELCOR Simulations of the Severe Accident at Fukushima Unit 3," Sandia National Laboratories, SAND2013-6886J, 2013, p.26.

[42] For the zirconium-steam reaction, every 2.53 kilograms (kg) of zirconium and 1.0 kg of steam that reacts yields 0.112 kg of hydrogen and 16.44 megajoules of thermal energy. See Fumiya Tanabe, "Analyses of core melt and re-melt in the Fukushima Daiichi nuclear reactors," Journal of Nuclear Science and Technology, Volume 49, No. 1, January 2012, p. 18.

[43] In 2018, the average U.S. residential utility customer consumed 10,972 kilowatt-hours of electricity for the entire year (about 30 kilowatt-hours per day). See U.S. Energy Information Administration, "Frequently Asked Questions: How much electricity does an American home use?" (available at: https://www.eia.gov/tools/faqs/faq.php?id=97&t=3 : last visited on 09/23/20)

that it takes 22,590 kg of zirconium to chemically react with 8,930 kg of steam to yield 1,000 kg of hydrogen—as well as 146,785 MJ of thermal energy.[44]

Provided steam is available under severe accident conditions, the reactor essentially becomes a furnace burning zirconium and other core contents, generating large amounts of thermal energy. In some severe accident scenarios, the burning of zirconium in steam (the zirconium-steam reaction) alone may generate nearly as much thermal energy, over a brief period of time, as an operating coal-burning power plant rated at 250 megawatts electric (MWe).

In a time period of one minute, 6,780 kg of zirconium may react with 2,680 kg of steam to yield 300 kg of hydrogen and 44,035 MJ of thermal energy.[45] Converting 44,035 MJ of thermal energy to electricity with the efficiency of a power plant, about 33 percent, results in about 14,680 MJ of electrical energy, which is equal to about 4,080 kWh of electricity: enough electricity to power more than 135 average American homes for a full day.

The NRC should learn that under severe accident conditions, enough thermal energy to power 135 average American homes for a full day may swiftly enter the primary containment of a BWR Mark I unit.

The NRC's explanation of its basis for the venting capacity it stipulated in Order EA-13-109 is not based on sound science. The NRC's mistake is egregious: one may wonder how the US nuclear regulator could be so incompetent. How could the US nuclear regulator, which presumably bases its decisions on sound science, not pseudoscience, fail to realize that chemical reactions in a melting-down reactor generate vast quantities of thermal energy?

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[44] For the zirconium-steam reaction, every 2.53 kilograms (kg) of zirconium and 1.0 kg of steam that reacts yields 0.112 kg of hydrogen and 16.44 megajoules of thermal energy. See Fumiya Tanabe, "Analyses of core melt and re-melt in the Fukushima Daiichi nuclear reactors," Journal of Nuclear Science and Technology, Volume 49, No. 1, January 2012, p. 18.

[45] Flooding a melting-down reactor core with coolant water may generate from 5.0 to 10.0 kg of hydrogen per second. See Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In-Vessel Hydrogen Sources," p. 15.

III.C.2. The Thermal Energy Generated during the Flooding of a Melting-Down Reactor

The OECD Nuclear Energy Agency has reported that flooding a melting-down reactor core with coolant water may generate from 5.0 to 10.0 kilograms (kg) of hydrogen per second.[46] These values are provided for a pressurized water reactor ("PWR") with a power level of 900 megawatts electric (MWe). Flooding a melting-down U.S. BWR with the same power level—900 MWe—would generate at least as many kilograms of hydrogen per second. The amount of hydrogen generated during the flooding of a melting-down reactor core is primarily determined by how much steam is available within the core to react with the hot surfaces of the core's zirconium contents. A typical BWR core has roughly three times as much zirconium as a PWR core.[47] Unlike U.S. PWR cores, U.S. BWR cores contain boron carbide (B_4C) neutron absorber materials—a total of about 1,200 kg in a BWR core with a power level of 3,800 megawatts thermal.[48] For equivalent masses of boron carbide and zirconium reacted, the boron carbide-steam reaction generates about six times as much hydrogen as the zirconium-steam reaction.[49] Moreover, for equivalent masses reacted, the boron carbide-steam reaction generates from four to five times as much thermal energy as the zirconium-steam reaction.[50] This additional thermal energy serves to intensify the zirconium-steam reaction within the core, increasing hydrogen generation rates. A severe accident experiment—CORA-17—simulating the flooding of a melting-down BWR core generated more thermal energy and hydrogen than counterpart experiments simulating the flooding of melting-down PWR cores—CORA-12 and CORA-13. Analyses concluded that the larger amounts of thermal

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[46] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In- Vessel Hydrogen Sources," p. 15.

[47] A BWR core with a power level of 3,800 megawatts thermal (MWt) has about 76,000 kg of zirconium; whereas, a PWR core with a power level of 3,800 MWt has about 26,000 kg of zirconium. See International Atomic Energy Agency, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 10.

[48] International Atomic Energy Agency, "Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants," IAEA-TECDOC-1661, July 2011, p. 7.

[49] S. Hagen *et al*., "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," FZKA-5679, August 1996, p. i.

[50] *Id*.

energy and hydrogen generated in the CORA-17 experiment were due to the boron carbide-steam reaction.[51]

Rates of hydrogen generation of 5.0 to 10.0 kg per second during the flooding of a melting-down reactor core may last for a period longer than one minute.[52] The hydrogen is generated by the reactor's contents, primarily its zirconium, reacting with steam. For the zirconium-steam reaction, every 2.53 kg of zirconium and 1.0 kg of steam that reacts yields 0.112 kg of hydrogen and 16.44 megajoules (MJ) of thermal energy (heat).[53]

For a quick analysis, to illustrate that the NRC's post-Fukushima vents need a significantly higher capacity than Order EA-13-109 requires, let us suppose that during the flooding of a melting-down reactor core hydrogen generation rates may only reach as high as 5.0 kg per second, not as high as from 5.0 to 10.0 kg per second. For simplicity's sake, let us also suppose that the hydrogen is solely generated by the zirconium-steam reaction, that none comes from other core materials reacting with steam.

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[51] *Id*., p. 1.

[52] Severe accident experiments that simulate the flooding of a melting-down BWR core generated hydrogen earlier than counterpart experiments that simulated the flooding of melting-down PWR cores. See S. Hagen *et al*., "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," FZKA-5679, August 1996, p. 8. In the CORA-17 experiment, which simulated the flooding of a melting-down BWR core, extremely high hydrogen generation rates lasted longer than one minute. See S. Hagen *et al*., "Comparison of the Quench Experiments CORA-12, CORA-13, CORA-17," FZKA-5679, August 1996, Figure 25 on p. 42. See also S. Hagen *et al*., "Large Bundle BWR Test CORA-18: Test Results," FZKA 6031, April 1998, Figure 71 on p. 105. A paper discussing the hydrogen generation rates that may occur during the flooding of a melting-down reactor core of a PWR with a power level of 900 MWe states that generation rates may reach as high as 300 kg per minute. The paper also states that the zirconium-steam reaction may generate hydrogen at a rate of 5.0 kg per second. See E. Bachellerie *et al*., "Generic approach for designing and implementing a passive autocatalytic recombiner PAR-system in nuclear power plant containments," Nuclear Engineering and Design, Vol. 221, Nos. 1-3, April 2003, p. 158. Providing information about the AP1000, a PWR with a power level of 1,110 MWe, Westinghouse stated that flooding the AP1000 reactor with coolant water during a severe accident might generate as much as about 300 kg of hydrogen per minute. (Note that, in contrast to the OECD Nuclear Energy Agency, Westinghouse stated that generating about 300 kg of hydrogen per minute is considered the bounding case, claiming that such a rate is "an artificially rapid rate." Westinghouse opined that "typical" hydrogen generation rates would be about 100 kg of hydrogen per minute, during the flooding of the reactor under severe accident conditions.) See Westinghouse, "Part 3 of 3 Transmittal of Westinghouse Proprietary and Non-Proprietary Responses to NRC Requests for Additional for AP1000 Application for Design Certification," November 2002, (ADAMS Accession No. ML031670868).

[53] Fumiya Tanabe, "Analyses of core melt and re-melt in the Fukushima Daiichi nuclear reactors," Journal of Nuclear Science and Technology, Volume 49, No. 1, January 2012, p. 18.

Note that this analysis provides a low estimate of the maximum value of the thermal energy that may be generated during the flooding of a melting-down reactor core. Hydrogen generation rates from the zirconium-steam reaction alone may be greater than 5.0 kg per second; additional hydrogen is also generated by steam reacting with other reactor core contents, such as stainless steel and boron carbide, which generates additional thermal energy. Furthermore, this analysis does not include the thermal energy that would be generated by decay heating.

If 5.0 kg of hydrogen were generated per second, then 113 kg of zirconium would burn (oxidize) per second. The energy generated by the zirconium-steam reaction is 6.5 MJ per kg of zirconium reacted. Burning 113 kg of zirconium per second generates 734 MJ of thermal energy per second. In sum: flooding the melting-down core of a BWR with a power level of approximately 900 MWe may generate 734 MJ of thermal energy per second.

A joule per second is equal to a watt. Hence, 734 MJ of energy per second is equal to a power level of 734 megawatts thermal (MWt). Such a power level may sustain for a period longer than one minute, during the flooding of a melting-down reactor core. (About one-third of the total thermal energy produced by a BWR is converted into electricity. Hence, a BWR with a power level of 900 megawatts electric (MWe) has a thermal power level of approximately 2,700 MWt.) A power level of 734 MWt is equal to 27 percent of the total thermal power of a reactor rated at 2,700 MWt. (Most BWR Mark I units operating in the United States have maximum power levels in the approximate range of 2,000 to 4,000 MWt.)

In order to maintain the primary containment of a BWR Mark I unit below its pressure limit at all times, during a severe accident, a hardened vent likely needs to have an immense capacity—perhaps the capacity to handle more than 25 percent of the reactor's rated maximum thermal power.

III.C.2.a. Plant Workers Might Flood a BWR Core without Knowing Its Actual Condition

Plant workers at Fukushima Daiichi suffered setbacks because they did not know the actual conditions inside the three stricken reactors as the accident progressed. Workers at a BWR plant are supposed to detect the start of a severe accident by measuring the water level in the reactor core—an unreliable method. At Fukushima Daiichi, workers were misled during the early hours of the accident by erroneous readings of the Unit 1 reactor's condition. The reactor's water level appeared to be above the top of the fuel rods when in fact they were uncovered and melting down. The water levels of all the stricken Fukushima reactors read erroneously high.[54]

BWR water level measurements may read erroneously high during low-pressure accidents or when temperatures in the drywell become extremely high. In general, water level measurements are unreliable once reactor core damage commences.[55]

Plant workers might flood a melting-down BWR core with coolant water without knowing its actual condition. Erroneous readings of the reactor's water level might mislead them into unintentionally flooding the melting-down core. If this were to occur, they would likely be unprepared for the immense quantities of thermal energy and noncondensable hydrogen gas that would rapidly enter the primary containment, overwhelming it and the hardened vents required by Order EA-13-109.

III.C.3. The Thermal Energy Generated if Molten Materials in the Reactor Relocate Downward and Vaporize Large Quantities of Water

During a severe accident, large amounts of thermal energy may be rapidly generated within a short period of time if molten materials in the reactor relocate downward to the lower region of the core and vaporize large quantities of water. The vaporized water (steam) reacts with the molten materials. A Sociotechnical Systems Safety Research Institute computer analysis of the Fukushima Daiichi accident estimated that large amounts of hydrogen and thermal energy were generated in a short time period—of unspecified duration—during the Unit 3 reactor meltdown. In the analysis, about 4.25 hours after the reactor core was uncovered by coolant water, a large amount of molten materials relocated downward and vaporized about 4,235 kg of water in the lower region

[54] Institute of Nuclear Power Operations, "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," INPO 11-005, November 2011, pp. 9, 10.

[55] International Atomic Energy Agency, "Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident," IAEA-TECDOC-955, August 1997, p. 26. And Institute of Nuclear Power Operations, "Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," INPO 11-005, November 2011, p. 10.

of the core, rapidly generating about 475 kg of hydrogen and 69,600 MJ of thermal energy.[56]

If molten materials relocate to the lower region of the core and vaporize large quantities of water, immense quantities of thermal energy and noncondensable hydrogen gas would rapidly enter the primary containment, overwhelming it and the hardened vents required by Order EA-13-109. Perhaps it was with scenarios like this in mind that Sandia National Laboratories, discussing filtered venting systems, opined that "it may be difficult to design vents that can handle the rapid transients involved [in a severe accident]."[57]

III.C.4. The Fission Chain Reaction (Criticality) May Recommence During a BWR Severe Accident

BWR cruciform-shaped neutron absorbers, "control rods," utilize stainless-steel-clad sheaths to contain boron carbide material. In a severe accident, when BWR control rods heat up to about 1,250°C (2,282°F), their stainless steel and boron carbide materials eutectically interact with one another and liquefy. The liquefied mixture relocates downward to the lower areas of the reactor core and pressure vessel. The fuel rods remain at least partly intact for a period of minutes after the control rods are no longer present. There is a time window in which the fission chain reaction (criticality) may recommence in the reactor if it is flooded with non-borated water.[58] If criticality recommences, a large blast of thermal energy is generated from fission, serving to heat up the reactor's contents, which generate additional thermal energy from the reaction of steam and the core's contents, primarily zirconium. A large amount of thermal energy (from both fission and chemical reactions) rapidly enters the primary containment, likely overwhelming it and the hardened vents required by Order EA-13-109.

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[56] Fumiya Tanabe, "Analyses of core melt and re-melt in the Fukushima Daiichi nuclear reactors," Journal of Nuclear Science and Technology, Volume 49, No. 1, January 2012, pp. 19, 22-24.

[57] Allen L. Camp *et al*., Sandia National Laboratories, "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, August 1983, (ADAMS Accession No. ML071620344), p. 2.66.

[58] Piotr Darnowski *et al*., "Investigation of the recriticality potential during reflooding phase of Fukushima Daiichi Unit-3 accident," Annals of Nuclear Energy, Vol. 99, 2017, p. 495. And P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 204.

In the Fukushima Daiichi accident, the Unit 3 reactor may have regained criticality after its control rods liquefied. Records show that plant workers added boron to freshwater and seawater that was injected into the Fukushima Daiichi reactors during the accident. However, it is likely that non-borated seawater was injected into the Unit 3 reactor over a period of 12 hours. An analysis found that the Unit 3 reactor had the potential to regain criticality if it was indeed injected with non-borated seawater.[59]

III.C.5. There Is No Guarantee the Vents the NRC Requires Would Prevent the Containment from Failing in a Severe Accident

It is perplexing that the NRC requires hardened containment vents for BWR Mark I units that are only expected to handle the thermal energy generated by the fuel's decay heating in the event of a severe accident. When the NRC stipulated design features for the vents, it seems to have overlooked the fact that immense amounts of thermal energy are generated by the reaction of steam and zirconium (as well as by the reaction of steam and other metals within the reactor) as a severe accident progresses. The NRC also seems to have overlooked the fact that flooding a melting-down reactor with coolant water may generate as much as 5.0 to 10.0 kilograms of noncondensable hydrogen gas per second,[60] which causes the primary containment to rapidly pressurize.

The hardened vents required by Order EA-13-109 likely would not prevent the primary containment from failing under severe accident conditions, as the containment heated and pressurized to levels beyond it and the vents' capacity. If the containment failed, hydrogen gas would enter the reactor building and explode, releasing large amounts of radioactive material into the environment.

[59] Piotr Darnowski *et al.*, "Investigation of the recriticality potential during reflooding phase of Fukushima Daiichi Unit-3 accident," Annals of Nuclear Energy, Vol. 99, 2017, pp. 495, 496.

[60] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the Analysis and Management of Accidents, "GAMA Perspective Statement on In- Vessel Hydrogen Sources," p. 15.

III.C.6. How the Containment Might Fail in a Severe Accident

A BWR Mark I primary containment might rupture violently if it were rapidly pressurized with hundreds of kilograms of hydrogen gas and heated with immense amounts of thermal energy, as occurs when a melting-down reactor is flooded with coolant water. However, if the containment's rates of internal pressurization were not too swift, the leakage of gases from containment penetrations might even prevent the containment from rupturing.[61] (Typical BWR containments have 175 penetrations, almost twice as many as typical PWR containments.[62])

The extreme pressures and temperatures—potentially greater than 315°C (600°F)—that may last a long time within the primary containment during a severe accident are expected to degrade non-metallic seals of the containment's penetrations (some penetration-seals might be degraded from aging prior to an accident), exacerbating the problem of the containment's leakage of explosive hydrogen gas.[63]

On March 18, 2011, a week after the Fukushima Daiichi accident initiated, when information about the accident was sparse, David Lochbaum, then-Director of the Union of Concerned Scientists' Nuclear Safety Project, posted a blog article, titled "Possible Cause of Reactor Building Explosions." He warned that, in a severe accident, an excessive pressure buildup within the primary containment has the potential to push the drywell head[64] upward and open gaps around the drywell head flange seal, providing leak pathways that allow hydrogen to escape into the reactor building. (Radionuclides would also escape.) He posited that this problem caused a lot of hydrogen to leak from primary containments at Fukushima Daiichi, leading to the explosions that devastated three

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[61] C. H. Hofmayer *et al*., "Containment Leakage During Severe Accident Conditions," BNL-NUREG-35286, CONF-8406124-13, 1984.

[62] NRC, "Regulatory Effectiveness Assessment of Option B of Appendix J," NUREG-1777, August 2003, (ADAMS Accession No. ML033030547), p. 2.

[63] C. H. Hofmayer *et al*., "Containment Leakage During Severe Accident Conditions," BNL-NUREG-35286, CONF-8406124-13, 1984.

[64] BWR Mark I drywell heads typically have diameters between 9 and 12 meters (30 and 40 feet). Drywell heads are removed (and reattached) once every 18 to 24 months to facilitate refueling when the reactor is shut down for a scheduled outage. See C. H. Hofmayer *et al*., "Containment Leakage During Severe Accident Conditions," BNL-NUREG-35286, CONF-8406124-13, 1984. See also Kevin R. Robb, "External Cooling of the BWR Mark I and II Drywell Head as a Potential Accident Mitigation Measure—Scoping Assessment," Oak Ridge National Laboratory, ORNL/TM-2017/457, August 2017, p. 1.

reactor buildings.[65] Onsite inspections of Fukushima's damaged reactor buildings conducted years after Lochbaum wrote his article indicate he was correct: elevated levels of radioactive contamination have been measured on shield plugs, located directly above drywell heads (indicating radionuclides leaked from open gaps around flange seals).[66]

Under severe accident conditions, the vents required by Order EA-13-109 do not have the capacity to prevent the primary containment from failing. An internal pressure buildup would push the drywell head upward and open gaps around the drywell head flange seal, allowing hydrogen to leak into the reactor building and detonate.

IV. THE NRC NEEDS TO DETERMINE WHAT THE CAPACITY OF HARDENED CONTAINMENT VENTS MUST BE IN ORDER TO HANDLE THE MOST EXTREME SEVERE ACCIDENT CONDITIONS

The NRC needs to perform safety analyses to determine what the capacity of hardened vents for BWR Mark I containments must be in order to handle the total amount of thermal energy that would be generated over some periods of time during a severe accident. Such analyses need to focus on the periods of time in which the greatest amounts of thermal energy would be generated. As discussed in Sections III.C.2, III.C.3, and III.C.4 of this petition, there are some periods of time during a severe accident in which very large amounts of thermal energy are rapidly generated—far more than that solely generated by decay heating.

As stated in Section III.C of this petition, Idaho National Engineering Laboratory ("INEL") prepared a report for the NRC in the late 1980s, declaring that the thermal energy generated by the chemical reaction of steam and metals within the reactor, primarily zirconium, during a severe accident is "several times" as great as that generated

[65] David Lochbaum estimated that drywell head flange seals at Fukushima Daiichi began to fail once the internal pressures of primary containments increased to approximately 70 pounds per square inch gauge (psig), inducing extensive leaking, as had occurred in a containment pressure test conducted for a BWR Mark I in the 1970s. See David Lochbaum, "Possible Cause of Reactor Building Explosions," All Things Nuclear, March 18, 2011. (available at: https://allthingsnuclear.org/dlochbaum/possible-cause-of-reactor-building-explosions : last visited on 03/29/20)

[66] J. Rempe *et al.*, "U.S. Efforts in Support of Examinations at Fukushima Daiichi—2017 Evaluations," ANL/LWRS-17/02, August 2017, pp. 42, 44, 65. And Kevin R. Robb, "External Cooling of the BWR Mark I and II Drywell Head as a Potential Accident Mitigation Measure—Scoping Assessment," Oak Ridge National Laboratory, ORNL/TM-2017/457, August 2017, p. 2.

by decay heating. The hardened containment vents the NRC currently requires—in accordance with Order EA-13-109—are only expected to handle a continuous thermal energy input at a rate equal to one percent of the reactor's rated maximum thermal power, which only accounts for the thermal energy generated by the fuel's decay heating,[67] while maintaining the primary containment at a pressure lower than its design pressure and pressure limit.[68]

INEL's late 1980s report maintained that a vent intended to remove heat from and depressurize a BWR Mark I primary containment under severe accident conditions should be able to handle a continuous thermal energy input at a rate equal to seven percent of the reactor's rated maximum thermal power.[69] Petitioners speculate that the safety analyses performed by INEL, as well as any possible safety analyses performed by others, informing INEL's conclusion about the needed capacity of hardened containment vents did not account for the periods of time in which the greatest amounts of thermal energy would be generated during a severe accident. For example, Petitioners speculate INEL did not consider the thermal energy that would be generated when flooding a melting-down reactor core with coolant water.

According to an OECD Nuclear Energy Agency report from 2001, computer safety models underestimate the extent of the zirconium-steam reaction that occurs during the flooding of a melting-down reactor core with coolant water.[70] Hence, such

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[67] NRC, "Compliance with Order EA-12-050, Reliable Hardened Containment Vents: Interim Staff Guidance," Attachment I, August 29, 2012, (ADAMS Accession No. ML12229A475), p. 5.

[68] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321), Attachment 2: "Requirements for Reliable Hardened Vent Systems Capable of Operation under Severe Accident Conditions at Boiling-Water Reactor Facilities with Mark I and Mark II Containments," p. 2.

[69] This value is based on calculations performed for the two BWR Mark I units at Peach Bottom Atomic Power Station, which at the time were both rated at 3,293 megawatts thermal (MWt). See K. C. Wagner *et al*., "An Overview of BWR Mark-1 Containment Venting Risk Implications," Idaho National Engineering Laboratory, NUREG/CR-5225, June 1989, (ADAMS Accession No. ML101870670), p. 25. See also NRC, "Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 2," October 18, 1994, (ADAMS Accession No. ML011490143), p. 1. Additionally, see NRC, "Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 3," July 18, 1995, (ADAMS Accession No. ML021580312), p. 1.

[70] Report by Nuclear Energy Agency Groups of Experts, OECD Nuclear Energy Agency, "In-Vessel and Ex-Vessel Hydrogen Sources," NEA/CSNI/R(2001)15, October 1, 2001, Part I: B. Clément (IPSN), K. Trambauer (GRS), and W. Scholtyssek (FZK), Working Group on the

models also underestimate the amount of thermal energy that is generated during flooding. Therefore, even if INEL did consider the thermal energy generated when flooding a melting-down reactor core, its safety analyses, as well as any possible safety analyses performed by others, would have underestimated the total amount of thermal energy that is actually generated during flooding.

As stated in Section III.C of this petition, Petitioners believe the capacity of the vents INEL recommended (capable of removing seven times as much thermal power as the NRC requires) would be incapable of handling the total amount of thermal energy that would be generated over some periods of time during a severe accident. In Section III.C.2 of this petition, Petitioners presented information indicating that hardened containment vents need to have the capacity to handle more than a continuous thermal energy input at a rate equal to seven percent of the reactor's rated maximum thermal power. Section III.C.2 implies that a vent likely needs to have an immense capacity—perhaps the capacity to handle more than 25 percent of the reactor's rated maximum thermal power.

As covered in Section III.C.2, flooding a melting-down reactor core with coolant water may generate 734 megajoules (MJ) of thermal energy per second—a power level of 734 megawatts thermal (MWt). (Such a power level may sustain for a period longer than one minute.) This is equal to 27 percent of the total thermal power of a reactor rated at 2,700 MWt. Moreover, large amounts of thermal energy may also be rapidly generated within a short time period in a scenario in which a large amount of molten materials in the reactor relocate downward to the lower region of the core and vaporize large quantities of water. Section III.C.3 of this petition discusses the findings of a computer analysis of the Fukushima Daiichi Unit 3 accident. In the analysis molten materials relocate downward to the lower core and vaporize about 4,235 kg of water, generating 69,600 MJ of thermal energy, from the reaction of steam and the molten materials, in a short time period (of unspecified duration).[71]

Analysis and Management of Accidents, "GAMA Perspective Statement on In- Vessel Hydrogen Sources," p. 9.

[71] Fumiya Tanabe, "Analyses of core melt and re-melt in the Fukushima Daiichi nuclear reactors," Journal of Nuclear Science and Technology, Volume 49, No. 1, January 2012, pp. 22-24.

In yet another scenario that produces a large amount of thermal energy: the fission chain reaction (criticality) may recommence during a severe accident, as discussed in Section III.C.4 of this petition, if the reactor is flooded with non-borated water after its neutron absorbers have liquefied and relocated downward to the base of the core or reactor pressure vessel.[72] If criticality recommences, a large blast of thermal energy is generated, which would rapidly enter the primary containment, likely overwhelming it and the hardened vents required by Order EA-13-109. It is possible that recriticality occurred during the Fukushima Daiichi Unit 3 reactor meltdown.[73]

Petitioners have identified a serious problem: the "reliable" hardened containment vents the NRC currently requires cannot be guaranteed to perform adequately in the event of a severe accident. The NRC issued Order EA-13-109 in the aftermath of the Fukushima Daiichi accident; however, it did not consider that in the event of a severe accident, hardened vents need to have the capacity to handle the total amount of thermal energy that would be generated by the chemical reaction of steam and metals within the reactor, primarily zirconium, as well as that generated by decay heating. It is highly likely that a severe accident at a BWR Mark I unit outfitted with the new vents would cause the primary containment to fail, releasing explosive hydrogen gas into the reactor building—just as occurred at three units during the Fukushima Daiichi accident.

High-capacity hardened vents need to be installed in BWR Mark I units—ones capable of preventing catastrophic rupturing of the containment as well as preventing the containment's drywell head from pushing upward and opening gaps around the drywell head flange seal, which releases hydrogen and radioactive material into the reactor building.

The time is past due for the NRC to live up to its motto, "Protecting people and the environment." The NRC needs to commit to determining what the capacity of hardened vents for BWR Mark I containments must be in order to handle the total amount of thermal energy that would be generated over some periods of time during a

[72] Piotr Darnowski *et al.*, "Investigation of the recriticality potential during reflooding phase of Fukushima Daiichi Unit-3 accident," Annals of Nuclear Energy, Vol. 99, 2017, p. 495. And P. Hofmann, "Current Knowledge on Core Degradation Phenomena, a Review," Journal of Nuclear Materials, 270, 1999, p. 204.

[73] Piotr Darnowski *et al.*, "Investigation of the recriticality potential during reflooding phase of Fukushima Daiichi Unit-3 accident," Annals of Nuclear Energy, Vol. 99, 2017, pp. 495, 496.

severe accident. Analyses need to focus on the periods of time in which the greatest amounts of thermal energy would be generated, which, as Section III.C.2 of this petition indicates, may reach higher than 25 percent of the reactor's rated maximum thermal power.

V. PETITIONERS' REQUEST IS CONSISTENT WITH ANOTHER NRC REGULATION REGARDING REACTOR CONTAINMENTS

Petitioners' request that the hardened containment vents required by Order EA-13-109 be capable of discharging the total amount of thermal energy that would be generated over some periods of time during a severe accident is consistent with an aspect of the agency's regulatory position for how reactor containment structures must perform under design-basis accident conditions. Criterion 50, "Containment Design Basis," of Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants," requires consideration of the energy that would be generated by chemical reactions during design basis accidents.

Criterion 50, "Containment Design Basis," states:

> The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 *energy from metal-water and other chemical reactions* that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters [emphasis added].[74]

Criterion 50, "Containment Design Basis," requires consideration of the total amount of thermal energy that would be generated during a design-basis accident, including that generated from the zirconium-steam reaction and other chemical reactions. In contrast,

[74] NRC, Criterion 50 of Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants." (available at: https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appa.html : last visited on 07/23/20)

the stipulations for hardened containment vents set forth in Order EA-13-109 do not require that the vents have the capacity to handle the total amount of thermal energy that would be generated during a severe accident.

Order EA-13-109 stipulated that the new BWR Mark I (and Mark II) vents must have the capacity to handle a continuous thermal energy input at a rate equal to one percent of the reactor's rated maximum thermal power, while maintaining the primary containment at a pressure lower than its design pressure and pressure limit.[75] The NRC has explained that this requirement of Order EA-13-109 only accounts for the decay heating that would occur during a severe accident.

On Order EA-12-050 (which also pertains to Order EA-13-109 because the vents required by both orders have the same capacity), the NRC states:

> The NRC staff has determined that, for a vent sized under conditions of constant heat input at a rate equal to 1 percent of rated thermal power and containment pressure equal to the lower of the primary containment design pressure and the PCPL [primary containment pressure limit], the exhaust-flow through the vent would be sufficient to prevent the containment pressure from increasing. *This determination is based on studies that have shown that the torus suppression capacity is typically sufficient to absorb the decay heat generated during at least the first three hours following the shutdown of the reactor with suppression pool as the source of injection, that decay heat is typically less than 1 percent of rated thermal power three hours following shutdown of the reactor, and that decay heat continues to decrease to well under 1 percent, thereafter.* Licensees shall have an auditable engineering basis for the decay heat absorbing capacity of their suppression pools, selection of venting pressure such that the HCVS [hardened containment vent system] will have sufficient venting capacity under such conditions to maintain containment pressure at or below the primary containment design pressure and the PCPL. If required, venting capacity shall be increased to an appropriate level commensurate with the licensee's venting strategy. Licensees may also use a venting capacity sized under conditions of constant heat input at a rate lower than 1 percent of thermal power if it can be justified by analysis that primary

[75] NRC, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions," EA-13-109, June 6, 2013, (ADAMS Accession No. ML13143A321), Attachment 2: "Requirements for Reliable Hardened Vent Systems Capable of Operation under Severe Accident Conditions at Boiling-Water Reactor Facilities with Mark I and Mark II Containments," p. 2.

> containment design pressure and the PCPL would not be exceeded [emphasis added].[76]

Order EA-13-109 was issued in the aftermath of the Fukushima Daiichi accident, in which three reactors melted down, generating large amounts of thermal energy from the zirconium-steam reaction and other chemical reactions (in addition to that generated by decay heating). It is perplexing that the NRC issued Order EA-13-109 without considering the fact hardened containment vents must have the capacity to handle the total amount of thermal energy that would be generated during a severe accident.

VI. WHY THIS PETITION IS NOT RE-LITIGATING PRIOR NRC DECISIONS

10 C.F.R. § 2.206 and NRC Management Directive 8.11 do not allow petitioners to use petitions for enforcement actions in order to essentially appeal or re-litigate prior NRC decisions. The NRC arguably decided BWR hardened containment vent matters in the past with the issuances of Generic Letter 89-16, Order EA-12-050, and Order EA-13-109; however, this petition neither appeals nor re-litigates those decisions for the following reasons:

1) Petitioners were not afforded the opportunity to review and comment on the proposed issuances of Generic Letter 89-16, Order EA-12-050, and Order EA-13-109. Consequently, Petitioners were denied opportunities to legally intervene in the scope and content of the NRC's requirements imposed by Generic Letter 89-16, Order EA-12-050, and Order EA-13-109.

2) Petitioners reviewed Generic Letter 89-16, Order EA-12-050, Order EA-13-109, and numerous related documents yet have not identified any evidence or reason to believe the NRC considered the need for hardened containment vents to have the capacity to handle the total amount of thermal energy that would be generated over some periods of time during a severe accident.

3) Petitioners reviewed Generic Letter 89-16, Order EA-12-050, Order EA-13-109, and numerous related documents yet have not identified any evidence or reason to

[76] NRC, "Compliance with Order EA-12-050, Reliable Hardened Containment Vents: Interim Staff Guidance," Attachment I, August 29, 2012, (ADAMS Accession No. ML12229A475), pp. 5-6.

believe the NRC addressed the fact that Idaho National Engineering Laboratory ("INEL")—in research performed for the NRC in the late 1980s—determined that to adequately perform in the event of a severe accident hardened containment vents need to have the capacity to handle seven percent of the reactor's rated maximum thermal power.[77] (As explained in this petition, Petitioners believe that even the vents INEL proposed would be incapable of handling the total amount of thermal energy that would be generated over some periods of time during a severe accident.)

4) Petitioners have identified two documents, "Compliance with Order EA-12-050, Reliable Hardened Containment Vents: Interim Staff Guidance," Attachment I[78] (which also pertains to Order EA-13-109 because the vents required by both orders have the same capacity) and "Basis for Venting Capacity in Order EA-13-109, 'Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions'," [79] which describe how the NRC staff determined that hardened containment vents must have the capacity to handle one percent of the reactor's rated maximum thermal power. These documents explain that the NRC's "determination is based on studies that have shown that the torus suppression capacity is typically sufficient to absorb the decay heat generated during at least the first three hours following the shutdown of the reactor with suppression pool as the source of injection, that decay heat is typically less than 1 percent of rated thermal power three hours following shutdown of the reactor, and that decay heat continues to decrease to well under 1 percent, thereafter."[80]

The NRC determined that hardened containment vents only need to handle the thermal energy that would be generated by decay heating during a severe accident. The NRC did not consider that hardened containment vents actually need to have the capacity

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[77] K. C. Wagner et al., "An Overview of BWR Mark-1 Containment Venting Risk Implications," Idaho National Engineering Laboratory, NUREG/CR-5225, June 1989, (ADAMS Accession No. ML101870670), p. 25.

[78] NRC, "Compliance with Order EA-12-050, Reliable Hardened Containment Vents: Interim Staff Guidance," Attachment I, August 29, 2012, (ADAMS Accession No. ML12229A475).

[79] NRC, "Basis for Venting Capacity in Order EA-13-109, 'Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions'," August 9, 2013, (ADAMS Accession No. ML13221A011).

[80] NRC, "Compliance with Order EA-12-050, Reliable Hardened Containment Vents: Interim Staff Guidance," Attachment I, August 29, 2012, (ADAMS Accession No. ML12229A475), p. 5.

to handle the total amount of thermal energy that would be generated over some periods of time during a severe accident.

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Deprived of formal opportunities in the past to contest or challenge the technical basis for the BWR hardened containment vent capacities imposed by the NRC when issuing Generic Letter 89-16, Order EA-12-050, and Order EA-13-109, Petitioners seek remedy by means of 10 C.F.R. § 2.206.

An NRC regulation, 10 C.F.R. § 50.100, "Revocation, Suspension, Modification, Amendment of Licenses and Construction Permits, Emergency Operations by the Commission," allows the NRC to re-visit reactor licensing decisions if information not considered in the original licensing decisions becomes known. Despite having issued operating licenses for BWR units, the NRC imposed additional requirements for hardened containment vents by means of Generic Letter 89-16, Order EA-12-050, and Order EA-13-109, based on information that was subsequently developed. This petition seeks to have the NRC impose additional requirements in order to provide reasonable assurance that the hardened containment vents mandated to mitigate severe accidents will have the capacity to satisfactorily fulfill that important role.

VII. CONCLUSION

Petitioners first request that the NRC suspend the operating licensees of BWR Mark I units until the hardened containment vents presently installed in such units are replaced with vents that are capable of discharging the total amount of thermal energy that would be generated over some periods of time during a severe accident. Secondly, Petitioners request that the NRC revoke the licenses of any BWR Mark I units that are not modified to fulfill Petitioners' first request.

To uphold its congressional mandate to protect the lives, property, and environment of the people of the United States, the NRC must not allow BWR Mark I units to operate with inadequate hardened containment vents. If implemented, the enforcement action proposed in this petition would help improve public and plant worker safety.

To: Margaret Doane
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Respectfully submitted,

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