UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 29, 2021

Mr. Paul Gunter
Director Reactor Oversight Project
Beyond Nuclear
7304 Carroll Avenue, #182
Takoma Park, MD 20912

Dear Mr. Gunter:

Your petition, on behalf of Mark Leyse and Beyond Nuclear, dated October 16, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20294A240), addressed to the U.S. Nuclear Regulatory Commission (NRC) Executive Director for Operations, has been referred to the Office of Nuclear Reactor Regulation for review in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.206, "Requests for action under this subpart." In the petition, you requested that the NRC take enforcement action against licensees of all boiling-water reactors (BWRs) with Mark I containment systems in the form of a suspension of operating licenses until all hardened containment vent systems (HCVS) are replaced. The bases for your request are summarized below:

1. The vents the NRC requires would not prevent a BWR Mark I primary containment from failing in a Severe Accident (SA).

2. The NRC's explanation of the basis for the capacity of the vents required by NRC Order EA-13-109, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated June 6, 2013 (ADAMS Accession No. ML13143A334), does not address SA chemical reactions as sources of heat addition.

3. Thermal energy is generated during the flooding of a melting-down reactor.

4. Plant workers might flood a BWR core without knowing its actual condition.

5. Thermal energy will be generated if molten materials in the reactor relocate downward and vaporize large quantities of water.

6. The fission chain reaction (criticality) may recommence during a BWR SA.

7. There is no guarantee the vents the NRC requires would prevent the containment from failing in an SA.

Consistent with NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions" (ADAMS Accession No. ML18296A043), a petition review board (PRB) was established to evaluate your petition. The PRB consists of staff from NRC headquarters who are

knowledgeable of reactor containment technology and the specific containment vent requirements in NRC Order EA-13-109. In evaluating your petition, the PRB collaborated on reviews of the NRC's records regarding the issues you raised in your petition.

The PRB's initial assessment was that your submittal does not meet the criteria in Management Directive 8.11, Directive Handbook Section III.C.1(b)(ii) for accepting petitions because the technical issues raised in your petition have already been the subject of NRC staff review and evaluation and none of the additional Section III.C.1(b)(ii) circumstances apply.

On December 18, 2020, you were informed by e-mail of the PRB's initial assessment (ADAMS Accession No. ML20356A013). On February 3, 2021, a public teleconference was held (transcript found at ADAMS Accession No. ML21048A057) between you and the PRB for you to clarify your concerns and provide any additional information you wished the PRB to consider before making a final determination. You provided a written response to support the teleconference (ADAMS Accession No. ML21035A173). In these documents related to the February 3, 2021, meeting you provided the following additional concerns to justify the requested enforcement actions in the petition:

1. The Analyses of NRC's NUREG-2206, "Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments," dated March 2018 (ADAMS Accession No. ML18065A048) did not simulate phenomena as harsh as the ones that occurred in the Fukushima Dai-ichi Accident including scenarios in which tens of thousands of megajoules of thermal energy were generated in a short time period.

2. NUREG-2206 analyses simulations included unrealistic responses because instrumentation was considered operational when it may not be in an SA.

3. The analyses of NUREG-2206 are not legally binding.

4. Is the NRC really claiming the problem of Mark I primary containment over-pressurization under severe accident conditions is finally solved?

The additional concerns from that meeting have been considered in the PRB's final determination regarding whether the petition meets the criteria for consideration under 10 CFR 2.206.

The PRB's final determination is unchanged in that the BWR Mark I HCVS bases described in your petition do not meet the criteria for consideration under 10 CFR 2.206 because these issues have previously been the subject of NRC staff review. In issuing NRC Order EA-13-109 and in subsequently issued documents described below, the NRC did consider and/or analyze and document impacts from the SA conditions included in your petition as described in the enclosure to this letter.

It is important to note that the HCVS installed in Mark I containments following the issuance of Generic Letter 89-16, "Installation of a Hardened Wetwell Vent" (ADAMS Accession No. ML031140220), dated September 1,1989, and subsequently upgraded in response to lessons learned from the 2011 Fukushima Dai-ichi accident were not intended to eliminate the risk of SAs and their consequences. Rather, the HCVS is an integral piece of a comprehensive approach to prevent core damage and mitigate the potential for uncontrolled radioactive releases in a significant number of accident scenarios. In addition to the HCVS hardware, this

comprehensive approach also consists of operator actions that are governed by symptom-based procedures.

Severe accident phenomena, including those described in your petition, have been extensively modeled by the NRC using a fully integrated, engineering-level computer code developed by Sandia National Laboratories. The results of these evaluations demonstrate that a reliable HCVS, in conjunction with procedures and corresponding operator actions, can prevent containment failure for a significant number of SA scenarios. However, there are also SA scenarios that were evaluated by the NRC where the HCVS and the procedural mitigative actions of the comprehensive approach could not be successfully implemented. In these cases, the containment could eventually fail.

As documented in several papers to the Commission, including those associated with SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities," dated June 18, 2015 (ADAMS Accession No. ML15022A218) and SECY-16-0041, "Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, and Enhanced Instrumentation," dated March 31, 2016 (ADAMS Accession No. ML16049A088), the risk associated with these SA scenarios that result in containment failure is below the backfit criteria (with significant margin) that the NRC would have to satisfy in order to require additional improvements to the HCVS. A more detailed description of the SA modeling and backfit evaluation is provided in the attachment.

The regulations in 10 CFR 2.206 provide an opportunity for the public to petition the NRC to take enforcement-related action, and, while the PRB determined that the issues raised do not require further review, the NRC understands that this process takes time, resources, and energy by petitioners. Accordingly, I thank you for taking the time to raise your concerns to the attention of the NRC and participating in this process.

Sincerely,

Caroline L. Carusone

Digitally signed by Caroline L. Carusone
Date: 2021.03.29 16:24:16 -04'00'

Brian W. Smith, Deputy Director
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Enclosure:
Response to Petition

cc: Paul Gunter, Director
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Response to Petition Regarding Severe Accident Capabilities of
Hardened Containment Vent System

The purpose of this enclosure is to provide the U.S. Nuclear Regulatory Commission (NRC) response to the petition, as supplemented, and specifically to each of the seven individual bases identified in the petition dated October 16, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20294A240), and each of the four individual concerns in the supplemental information discussed by the petitioner during the teleconference on February 3, 2021.

General Response to the October 16, 2020, Petition

On December 18, 2020 (ADAMS Accession No. ML20356A013), the petitioner was informed by e-mail of the Petition Review Board's (PRB) initial assessment of the October 16, 2020, petition. The information provided below is consistent with the response provided in the December 18, 2020, e-mail.

The hardened containment vent system (HCVS) is one of a range of tools that the operators could use to prevent or mitigate an environmental release. The purpose of the HCVS is not to prevent containment failure without the mitigating strategies discussed in Order EA 13-109. This is documented in Section II of NRC Order EA 13-109 (ADAMS Accession No. ML13143A334) as shown, in part below:

> On March 12, 2012, the NRC issued Order EA-12-050 ["Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents," March 12, 2012, ADAMS Accession No. ML12054A694.] requiring the Licensees identified in Attachment 1 to this Order to implement requirements for a reliable hardened containment venting system (HCVS) for Mark I and Mark II containments. Order EA-12-050 required licensees of BWR [boiling-water reactor] facilities with Mark I and Mark II containments to install a reliable HCVS to support strategies for controlling containment pressure and preventing core damage following an event that causes a loss of heat removal systems (e.g., an extended loss of electrical power).
>
> The requirements in this Order, in addition to providing a reliable HCVS to assist in preventing core damage when heat removal capability is lost (the purpose of EA-12-050), will ensure that venting functions are also available during severe accident conditions. Severe accident conditions include 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.
>
> Ensuring that the venting functions are available under severe accident conditions will support the strategies in the Mark I and Mark II severe accident management guidelines for the protection or recovery of the containment, which serves as a barrier to the release of radioactive materials. This Order will ensure that this additional severe accident venting capability is provided while also achieving, with minimal delays, the purpose of EA-12-050 — to provide a reliable

HCVS to control containment pressure and prevent core damage following the loss of heat removal functions.

For accidents that are beyond the design basis, strategies for controlling containment pressure and preventing core damage include operators using the emergency operating procedures or severe accident (SA) management guidelines (SAMGs), as appropriate, to prevent or mitigate a release. The SAMGs include mitigation strategies that operators can implement such as SA water addition (SAWA) and SA water management (SAWM) and lists of equipment needed to employ those strategies. The operator would decide what equipment to use based on its availability and its potential effect on an environmental release.

To explore potential enhancements to measure for preventing and mitigating a release, the NRC performed a study documented in NUREG-2206, “Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments,” dated March 2018 (ADAMS Accession No. ML18065A048). The study applied the NRC’s fully integrated, engineering-level MELCOR Computer Code which was developed by Sandia National Laboratories to simulate accident progression and source term for a range of SA scenarios. The MELCOR code models SA phenomena that could contribute to containment pressurization, including:

- steam oxidation of zirconium and steel producing hydrogen and heat;
- steam oxidation of boron carbide producing heat, hydrogen, and carbon monoxide;
- reflooding of an uncovered core producing steam;
- core debris relocation into residual water in the reactor vessel lower plenum producing steam;
- core debris relocation onto the containment floor producing steam, hydrogen, and carbon monoxide by corium-concrete interaction.

The study considered a range of mitigation strategies including venting with the HCVS, SAWA and SAWM. The study demonstrated that the use of HCVS was one of several beneficial strategies for preventing and mitigating releases and concluded that additional regulatory requirements were not justified.

The NRC staff is aware of the potential for recriticality as a result of relocation of control rods from the core and reflooding the core with unborated water. However, the NRC staff does not believe that this issue affects the conclusions of the NUREG-2206 study, because (1) the limited time period for simultaneously having relocated control rods and intact fuel assemblies and (2) the limited likelihood of having a core-wide occurrence of relocated control rods and intact fuel assemblies. NRC Generic Safety Issue (GSI) 155, “Generic Concerns Arising from TMI-2 [Three Mile Island, Unit 2] Cleanup,” Revision 3 (ADAMS Accession No. ML11353A382), considers the potential for recriticality in BWRs during SA conditions, which is discussed on page 3.155-5. Recriticality would require a core reflood occurring specifically after all control rods have melted but prior to the occurrence of significant fuel rod melting. This is a very narrow window in an SA event. Due to the extreme unlikelihood of a recriticality event occurring, the event is not explicitly modeled in the analysis documented in SECY-12-0157, “Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments (REDACTED VERSION),” dated November 26, 2012 (ADAMS Accession No. ML12345A030), that supports the basis for the SA capabilities of the HCVS. Similarly, because of the very remote chance of a recriticality event, the NRC

Order EA-13-109, Attachment 2, Section 1.2.1 requirement which specifies that HCVS be designed for 1 percent of licensed thermal power, does not consider such an event.

The following numbered paragraphs correspond to each of the 11 bases or concerns provided both in the original petition dated October 16, 2020, and during the February 3, 2021, meeting. Each is listed below and is followed by the corresponding past NRC consideration and/or review of that basis or concern.

Responses to Specific Bases Identified in the October 16, 2020, Petition

1. The Vents the NRC Requires Would Not Prevent a BWR Mark I Primary Containment from Failing in an SA.

NRC Response: NRC Order EA-13-109 requires installation of a reliable HCVS that will not only assist in preventing core damage when heat removal capability is lost, but will also function in SA conditions (i.e., when core damage has occurred). The HCVS is one of a range of tools that the operators could use to prevent or mitigate an environmental release. The purpose of the hardened containment vent system (HCVS) is not to prevent containment failure without the mitigating strategies discussed in Order EA 13-109. Order EA-13-109 venting upgrades are intended to increase confidence in preventing an uncontrolled release following core damage events.

2. "The NRC's Explanation of the Basis for the Capacity of the Vents Required by Order EA-13-109" does not Address SA Chemical Reactions as Sources of Heat Addition.

NRC Response: In NUREG-2206, the NRC considered all the phases of SAs, from heat up of the core to fuel degradation and debris relocation to the lower plenum and finally failure of the reactor lower head. NUREG-2206 assumed that the HCVS required by Order EA-13-109 was in place. In considering all phases of an SA, by definition, NUREG-2206 explicitly discusses hydrogen generation and molten core concrete interactions that result from core degradation in an SA. Section 3, "Accident Progression Analysis," of NUREG-2206 provides a detailed discussion of the plant model including the boundary conditions (e.g., operator control, SAWA), the run matrix, and the results of accident progression (see Figure 3-14 for snapshots of core degradation process until lower head failure). For example, Table 3-4 contains the total hydrogen generation in the core and Figure 3-16 shows the post core damage vent operation and containment pressure response.

3. The Thermal Energy is Generated During the Flooding of a Melting-Down Reactor.

NRC Response: The MELCOR SA progression in NUREG-2206 considers all material oxidation processes including boron carbide (B4C) neutron absorber materials by steam and hydrogen generation upon failure of the control blade sheaths. The details of the B4C model can be found in "MELCOR Computer Code Manuals, Volume 2: Reference Manual, Version 2.2.9541 2017," SAND 2017-0876 O, Sandia National Laboratories, dated January 2017 (ADAMS Accession No. ML17040A420). This is discussed in Section 2.5, "Oxidation," page COR-RM-83 of the reference manual.

4. Plant Workers Might Flood a BWR Core Without Knowing its Actual Condition.

NRC Response: In NUREG-2206, the NRC considered a sensitivity calculation with water addition shortly after core heat up and prior to vessel breach when the water level reaches the

bottom of active fuel. In this scenario, core cooling led to a delay in the timing of containment venting and significantly reduced the amount of radionuclide release to the environment. This sensitivity calculation is discussed on pages 3-56 through 3-58 of NUREG-2206.

5. Thermal Energy will be Generated if Molten Materials in the Reactor Relocate Downward and Vaporize Large Quantities of Water.

NRC Response: Relocation of debris to the lower plenum is explicitly modeled in the code calculations documented in NUREG-2206. This phenomenon leads to transfer of energy from the hot debris to the water present in the lower plenum resulting in boiling of the water and in some cases to a pressure spike inside the reactor pressure vessel. However, the containment is successfully vented in this scenario. An example of this phenomenon can be found in Figure 3-10 of NUREG-2206.

6. The Fission Chain Reaction (Criticality) May Recommence During a BWR SA.

NRC Response: GSI 155 includes a potential for recriticality in BWRs during SA conditions, which is discussed on page 3.155-5. Recriticality would require a core reflood occurring specifically after all control rods have melted but prior to the occurrence of significant fuel rod melting. This is a very narrow window timewise that operators are aware of. Due to this extreme unlikelihood of a recriticality event occurring, the event was not explicitly modeled in the analysis documented in SECY-12-0157 that supported the basis for the SA capabilities of the HCVS including the NRC Order EA-13-109, Attachment 2 Section 1.2.1 requirement that provides the requirement that the HCVS be designed for 1 percent of licensed thermal power. Subsequent analysis documented in NUREG-2206 and SECY 16-0041, “Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control and Enhanced Instrumentation,” dated March 31, 2016 (ADAMS Accession No. ML16049A088), modeled several SA scenarios and demonstrates that additional changes to the HCVS to accommodate a higher thermal power meet neither the adequate protection threshold nor the cost justified substantial safety benefit requirements for ordering a change to the HCVS design.

7. There Is No Guarantee the Vents the NRC Requires Would Prevent the Containment from Failing in an SA.

NRC Response: There is no guarantee that a reactor core containment will never fail. The NRC regulatory structure is intended to provide reasonable assurance of adequate protection of public health and safety, including consideration of unlikely events such as containment failures.

Enclosure 1 of SECY 16-0041 (ADAMS Accession No. ML16049A291) specifically considers pressure and temperatures associated with SA phenomena and provides reasonable assurance that containment failure can be prevented when an SA occurs.

In NUREG-2206, the NRC concludes that venting alone is not necessarily an adequate means of preventing containment failure as venting does not prevent other modes of containment failure (liner melt-through and over-temperature failure of the upper drywell head that can lead to the bypass of the suppression pool and direct release of radioactivity to the environment). A combination of venting and SAWA/SAWM is required to prevent such failures.

Responses to Concerns Identified in the February 3, 2021, Teleconference

On February 3, 2021, a public teleconference was held between the petitioner and the petition review board (PRB) (transcript found at ADAMS Accession No. ML21048A057) for the petitioner to clarify its concerns or provide any additional information for the PRB to consider before making a final determination. The petitioner also provided a written response to support the teleconference (ADAMS Accession No. ML21035A173).

1. The Analyses of NUREG-2206 did not Simulate Phenomena as Harsh as the Ones That Occurred in the Fukushima Dai-ichi Accident Including Scenarios in which Tens of Thousands of Megajoules of Thermal Energy were Generated in a Short Time Period.

The SA phenomena described in your petition have been extensively modeled by the NRC as documented in NUREG-2206, as well as other analyses performed by the NRC staff that predate the Fukushima Dai-ichi accident. This includes the State-of-the-Art Reactor Consequence Analyses (SOARCA) project that developed best estimates of the offsite radiological health consequences for potential severe reactor accidents. The SOARCA analyzed the potential consequences of SAs at the Surry Power Station near Surry, Virginia, and the Peach Bottom Atomic Power Station (Peach Bottom) near Delta, Pennsylvania. Peach Bottom has a Mark I containment. The SOARCA project, which began in 2007, combined up-to-date information about the plants' layouts and operations with local population data and emergency preparedness plans. This information was then analyzed using state-of-the-art computer codes that incorporated decades of research into severe reactor accidents. Supporting technical information regarding the Peach Bottom analysis is available in NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project," Volume 1 "Peach Bottom Integrated Analysis," dated May 2013 (ADAMS Accession No. ML13150A053).

The analysis documented in NUREG-2206 used the SA modeling tools that were also used in the SOARCA project and include the phenomena identified by the petition. As described above, the sequence of the SA is an important aspect of how the NRC has modeled SA progression in Mark I containments for decades. The scenarios described in the petition assume that the energy released from the phenomena you identify must be instantaneously passed through the HCVS in order to prevent containment failure. This assumption does not consider other means of heat transfer (e.g., heat needed to melt the fuel in an SA as well as heat that is removed to the surrounding structures).

In the case where use of the HCVS is employed, in combination with successful deployment of SAWA/SAWM strategies, containment does not fail. As noted above, the NRC has concluded that venting alone is not necessarily an adequate means to prevent containment failure as venting does not prevent other modes of containment failure (liner melt-through and over-temperature failure of the upper drywell head that can lead to the bypass of the suppression pool and release of radioactivity to the environment). A combination of venting and SAWA/SAWM is required to prevent such failures.

The NRC required all U.S. nuclear power plants, similar in design to the Fukushima plants, to improve or install a reliable, hardened vent to remove heat and control pressure before potential reactor core damage. The HCVS in combination with the mitigation strategies equipment required per 10 CFR 50.155, "Mitigation of beyond design basis events," allows decay heat to be removed from the reactor during a beyond-design-basis event extended loss of alternating

current power without fuel melting. Therefore, the mitigation strategies equipment results in the risk of reactor core damage from beyond-design-basis events, which is already low, being even lower.

The NRC considered SA progressions where use of the HCVS and SAWA/SAWM strategies are not successfully implemented. In these cases, the containment eventually fails. As documented in several papers to the Commission, including those associated with SECY-15-0085, “Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities,” dated June 18, 2015 (ADAMS Accession No. ML15022A218), and SECY-16-0041, the risk associated with the SA scenarios that result in containment failure is below the backfit criteria (with significant margin) that the NRC would have to meet in order to require additional improvements to the HCVS.

1a. Petitioner asserts that based on NUREG/CR-5225, Mark I Vents Need the Capability to Vent 7 percent of Rated Thermal Load

The petitioner asserts that analysis in NUREG/CR-5225, “An Overview of BWR Mark I Containment Venting Risk Implications” (ADAMS Accession No. ML101870670), states that in order to successfully operate under SA conditions a hardened containment vent needs to have a capacity that is seven times as great as that of the “reliable” vents required by the NRC.

The analyses documented in NUREG/CR-5225 note that, during certain SA scenarios, 18-inch lines, which can remove approximately 7 percent of the rated thermal core power, could adequately depressurize the system. The scenario that leads to needing this amount of energy removal is an anticipated transient without scram event,[1] The NUREG/CR-5225 report is not referring to an extended loss of alternating current power event that resulted in Order EA-13-109 being issued.

NUREG 2206 provides a history of the Mark I containment accident analyses that were performed since the 1980s including the SOARCA project to develop best estimates of the offsite radiological health consequences for potential severe reactor accidents, the post Fukushima Dai-ichi accident analysis that was performed to support requirements found in Order EA-13-109, and the analysis performed to support the containment protection and release reduction rulemaking. This information was then analyzed using the MELCOR Codes which incorporate decades of research into severe reactor accidents. The analyses, which were documented and referenced in NUREG-2206, include SECY-12-0157 and SECY-15-0085.

Subsequent analyses not referenced in NUREG-2206 associated with SA phenomena can be found in SECY-16-0041.

These post-Fukushima analyses confirmed the effectiveness of the measures required by Order EA-13-109 and concluded that further enhancements to the HCVS were not needed.

1b. NUREG-2206 was Completed After Issuing Order EA-13-109

Although NUREG-2206 was issued after Order EA 13 109, as discussed by the petitioner on February 3, 2021, NUREG-2206 captures the history of the multiple analyses that were performed to support both the requirements found in Order EA-13-109 regarding the HCVS capabilities and analyses that were subsequently performed to support the containment

1 See 10 CFR 50.62 for anticipated transient without scram information

protection and release reduction potential rulemaking activity. NUREG 2206 provides the history of the analyses including referencing the analyses that were documented in SECY-12-0157 and that were used to support issuance of Order EA-13-109. The analyses in SECY-12-0157 (as captured in NUREG-2206) emphasize, in particular, the functionality of the reliable hardened vents under SA conditions. NUREG-2206 also provides additional references to the containment protection and release reduction rulemaking activity documented in SECY-15-0085. NUREG 2206 documents the technical work that was done to support the regulatory basis for this potential rulemaking.

The NRC staff references NUREG-2206 in the response to the petition because it provides the history of the analyses that support both phases of Order EA-13-109 as well as the analyses that support the containment protection and release reduction rulemaking activity.

2. NUREG-2206 Analyses Simulations Included Unrealistic Responses Because Instrumentation was Considered Operational When it may not be in an SA.

The guidance for the qualification of instruments relied on for SAWA/SAWM strategies in accordance with Order EA-13-109, can be found in NEI 13-02, Revision 1, “Industry Guidance for Compliance with Order EA-13-109, BWR Mark I & II Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions,” dated April 2015 (ADAMS Accession No. ML15113B318), as endorsed by Japan Lessons-Learned Project Directorate (JLD) interim staff guidance (ISG) JLD-ISG-2013-02, “Compliance with Order EA- 13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operations under Severe Accident Conditions,” dated November 2013 (ADAMS Accession No. ML13304B836), and JLD-ISG-2015-01, “Compliance with Phase 2 of Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions,” dated April 2015 (ADAMS Accession No. ML15104A118).

The NRC staff assessed the instrumentation qualifications as part of the safety evaluations associated with the implementation of this Order. The individual safety evaluations can be accessed through the following Web site: https://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/japan-plants.html.

In addition to the guidance associated with Order EA-13-109, the NRC staff performed additional analyses associated with SA instrumentation as documented in SECY-16-0041, Enclosure 2, “Enhanced Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions” (ADAMS Accession No. ML16049A295). As described in this SECY paper, during review of the NRC Near-Term Task Force (NTTF) report recommendations (ADAMS Accession No. ML11186A950), the ACRS noted that Section 4.2 of the NTTF report discusses how the Fukushima operators faced significant challenges in understanding the condition of the reactors, containments, and spent fuel pools because the existing design basis instrumentation was either lacking electrical power or was providing erroneous readings. As a result, an additional recommendation was developed to address the regulatory basis for requiring reactor and containment instrumentation to withstand beyond design basis accident conditions.

The 21-page assessment of SA instrumentation can be found in SECY160041, Enclosure 2. The Enclosure discusses the guidelines used to validate readings to determine emergency operating procedures and SAMG implementation. The Enclosure also discusses the use of calculational aides to assist operators during an SA. SECY160041, Enclosure 2, concludes that further regulatory actions associated with “Enhanced Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions,” is not warranted.

3. The Analyses of NUREG-2206 are not Legally Binding.

The NRC staff developed the MELCOR analyses to simulate SAs (including the harsh conditions experienced at Fukushima) over a period of decades. MELCOR analyses in the NUREG demonstrate the efficacy of mitigation strategies that are imposed on licensees' operating plants with Mark I containments by Order EA-13-109, including the SA capabilities associated with this Order. As noted above, the analyses documented in NUREG-2206 as well as other analyses performed (e.g., SECY-16-0041) modeled both the success of the HCVS in combination with SAWA/SAWM strategies to maintain containment integrity and scenarios where containment fails. In the SA scenarios where containment fails, the NRC staff concluded that changes to Mark I and II containments beyond those required by Order EA-13-109 would not constitute substantial safety improvements.

4. Is the NRC Really Claiming the Problem of Mark I Primary Containment Over-Pressurization Under SA Conditions is Finally Solved?

After the Fukushima Dai-ichi accident, the NRC required all U.S. nuclear power plants similar in design to the Fukushima plants to improve or install a reliable, hardened vent to remove heat and pressure before potential reactor core damage. The HCVS in combination with the mitigation strategies equipment required per 10 CFR 50.155, allows decay heat to be removed from the reactor during a beyond-design-basis event extended loss of alternating current power thus preventing core damage. The HCVS capabilities in combination with the mitigation strategies equipment results in the risk of reactor core damage from beyond-design-basis events, which is already low, being even lower.

The capabilities of the HCVS required by Order EA-13-109, in addition to assisting in preventing core damage, provide mitigation capabilities in the unlikely event that core damage occurs. The HCVS SA capabilities were addressed in a phased approach in accordance with the Order. The HCVS along with the SAWA/SAWM strategies provide for additional SA mitigation beyond those which existed prior to the Fukushima-Daiichi accident.

The NRC Staff examinations of the HCVS capabilities required by Order EA-13-109 confirm the effectiveness of the measures required by the Order to both prevent an SA and to mitigate an SA should one occur, and concludes that further enhancements are not needed to this system.

SUBJECT: OEDO-20-00413 - 2.206 PETITION FOR BOILING WATER REACTOR HARDENED CONTAINMENT VENTS (EPID L-2020-CRS-0003) DATED MARCH 29, 2021

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DATE	3/3/2021	3/4/2021	3/3/2021	3/3/2021
OFFICE	NRR/DORL/LPL22/BC	OGC – NLO	NRR/DORL/DD	NRR/DANU/DD
NAME	UShoop	RCarpenter	CCarusone	BSmith
DATE	3/8/2021	3/17/2021	3/17/2021	3/17/2021
OFFICE	NRR/DD	NRR/DANU/DD		
NAME	MKing	BSmith (CCarusone for)		
DATE	3/29/2021	3/29/2021		

OFFICIAL RECORD COPY