Tam, Peter

From: saporito3@gmail.com on behalf of Thomas Saporito [thomas@saprodani-associates.com]
Sent: Thursday, April 14, 2011 2:59 PM
To: Tam, Peter
Subject: 10 CFR 2.206 Petition Documents
Attachments: SA20110414.17.pdf; SA20110414.13.pdf; SA20110414.14.pdf; SA20110414.15.pdf; SA20110414.16.pdf

Mr. Tam:

Attached please find a copy of the documents associated with our 10 CFR 2.206 petition filed with the NRC on March 12, 2011. Please provide a copy of the attached documents to the NRC Control Desk - per request of the NRC PRB Chairman.

Kind regards,
Thomas Saporito

Thomas Saporito, Senior Consulting Associate
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Saprodani-Associates - Advocate/GreenPeace USA

SA20110414.17

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 20, 1972

J. F. O'Leary, L
F. E. Kruesi, RO
L. Rogers, RS

Here is an idea to kick around. Please let me have your reactions.

S. H. Hanauer, DRTA

cc: E. G. Case, L
J. M. Hendrie, L
D. F. Knuth, L
R. L. Tedesco, L
V. Stello, L
G. Lainas, L

DRAFT

Pressure-Suppression Containments

1. Conclusions and Recommendations

Recent events have highlighted the safety disadvantages of pressure-suppression containments. While they also have some safety advantages, on balance I believe the disadvantages are preponderant. I recommend that the AEC adopt a policy of discouraging further use of pressure-suppression containments, and that such designs not be accepted for construction permits filed after a date to be decided (say two years after the policy is adopted).

2. Discussion

A pressure-suppression containment system has some means of absorbing the heat of vaporization of the steam in the fluid released to the containment volume. In all three GE models, the steam is forced to bubble through a pool of water and is condensed. In the Westinghouse design, the steam is condensed by flowing it over ice cubes. The objective is to reduce the pressure in the containment through "suppressing" the partial pressure of the steam by condensing it. To be effective, pressure suppression must take place concurrent with the flow of steam into the containment, and its effectiveness is therefore dependent on the rate at which steam is generated or released. If some unexpected event should result in steam generation or flow greater than the suppression capability, then the steam that is not condensed would add an increment of containment pressure. Since the objective of pressure suppression is to permit use of a smaller containment, rated at lower pressure than would be required without suppression, then incomplete suppression would lead to overpressurizing a pressure-suppression containment so designed.

It may be noted that the Stone and Webster "subatmospheric" design has little effect on the initial containment pressure rise due to an accident, and is therefore not a "pressure-suppression containment" for the present discussion. In this design, chilled water sprays are used to reduce the containment pressure, and therefore the containment leakage, quickly after a postulated LOCA. The pressure capability and volume are designed to take the full accident, without credit for condensation.

Like all containments, the pressure-suppression designs are required to include margins in capability. Experiments have been conducted by GE and Westinghouse to establish the rate of steam generation that can be accommodated. The pressure-suppression pools, ice condenser, etc., are then sized for the double-ended break steam flow, with margins for unequal distribution of steam to the many modular units of which the condenser is composed. The rate and distribution margins are probably adequate.

More difficult to assess is the margin needed when applying the experimental data to the reactor design. Recently we have reevaluated the 10-year-old GE test results, and decided on a more conservative interpretation than has been used all these years by GE (and accepted by us). We now believe that the former interpretation was incorrect, using data from tests not applicable to accident conditions.

We are requiring an independent evaluation of the ice condenser design and its bases to make less probable any comparable misinterpretation of this design.

Since the pressure-suppression containments are smaller than conventional "dry" containments, the same amount of hydrogen, formed in a postulated accident, would constitute a higher volume or weight percentage of the containment atmosphere. Therefore, such hydrogen generation tends to be a more serious problem in pressure-suppression containments. The small GE designs (both the light-bulb-and-doughnut and the over-under configurations) have to be inerted because the hydrogen assumed (per Safety Guide 7) would immediately form an explosive mixture. The GE Mod 3 and the Westinghouse ice condenser designs (they have equal volumes) require high-flow circulation and mixing systems to ensure even dilution of the hydrogen to avoid flammable mixtures in one or more compartments (see following for an additional serious disadvantage of this needed recirculation and its valves). By contrast, the dry containments only require recombination or purging starting weeks after the accident.

All pressure-suppression containments are divided into two (or more) major volumes, the steam flowing from one to the other through the condensing water or ice. Any steam that flows from one of these volumes to the other without being condensed is a potential source of unsuppressed pressure. Neither the strength nor the leakage rate of the divider (between the volumes) is tested in the currently approved programs for initial or periodic inservice testing. Some effort is now underway to devise a leakage test, but none has so far been accomplished.

Because of limited strength against collapse, the "receiving" volume has to be provided with vacuum relief. In all designs except GE Mod III, this function is performed by a group of valves. Such a valve stuck open is a large bypass of the condensation scheme; the amount of steam that thus escapes condensation can overpressurize the containment.

Valves do not have a very good reliability record. Recently, five of the vacuum relief valves for the pressure-suppression containment of Quad Cities 2 were found stuck partly open. Moreover, these valves had been modified to include redundant "valve-closed" position indicators and testing devices, because of recent Reg concerns. The redundant position indicators were found not to indicate correctly the particular partly open situation that obtained on the five failed valves. We have only recently begun to pay serious attention to these valves, so previous surveillance programs have not generally included them. The GE Mod III design has an elegant water-leg seal that obviates the need for vacuum relief valves.

The high-capacity atmosphere recirculation systems provided for hydrogen mixing involve additional valves which, if open at the wrong time, would constitute a serious steam bypass and thus a potential source of containment over-pressurization. These valves are large, and must open quickly and reliably when recirculation is needed. In other engineered safety features, no single valve is relied on for such service, yet redundancy has not been provided even for single failures, open and closed, of these valves. This is a serious mission, since opening at the wrong time leads to over-pressurization, while failure to open when needed inhibits recirculation.

The smaller size of the pressure-suppression containment, plus the requirement for the primary system to be contained in one of the two volumes, has led to overcrowding and limitation of access to reactor and primary system components for surveillance and in-service testing. Separate shielding of components has tended to subdivide into compartments the volume occupied by the primary system. (Some compartmentation of dry containments also occurs.) A pipe break in one of these compartments creates a pressure differential; each compartment must be designed to withstand this pressure. A method of testing such designs has not been developed.

What are the safety advantages of pressure suppression, apart from the cost saving. GE people talk about a decontamination factor of 30,000 from scrubbing of iodine out of the steam by the water. This is hard to swallow, but some decontamination undoubtedly occurs. One wonders why GE doesn't do an experiment to measure it, and get credit for it. The ice condenser decontamination is measurable but not significant.

Recirculation of the containment atmosphere through the ice has the potential for rapidly reducing the containment pressure by cooling its atmosphere. But in the present design there's not enough ice for that, ~~so containment~~ sprays are furnished (in both volumes), just as in dry containments. Recirculation through the water in the GE designs seems not to have been tried, but may be necessary in Mod 111 for hydrogen control. We have no analysis whether any significant cooling will result.

It is by no means clear that the pressure-suppression containments are, overall, significantly cheaper than dry containments when all costs are included. Information on this point would be useful in evaluating costs and benefits, and should be obtained.

Saprodani Associates

March 12th, 2011

Honorable William Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In re: Petition Under 10 C.F.R. 2.206 Seeking Enforcement Action Against Licensees of the U.S. Nuclear Regulatory Commission

Dear Mr. Borchardt:

Enclosed herewith, please find a Petition filed under 10 C.F.R. 2.206 seeking enforcement action against licensees of the U.S. Nuclear Regulatory Commission (NRC).

Sincerely,

Thomas Saporito

Thomas Saporito
Consulting Associate

Post Office Box 8413 • Jupiter • Florida 33468 • Phone: 561-972-8363 • Fax: 561-247-6404 • Email: saporito3@gmail.com

UNITED STATES NUCLEAR REGULATORY COMMISSION
BEFORE THE HON. WILLIAM BORCHARDT

In the Matter of:

SAPRODANI ASSOCIATES, **DATE: 12 MAR 2011**
Petitioners

v.

NUCLEAR REGULATORY COMMISSION,

and the

NUCLEAR INDUSTRY,
Respondents.

Docket Nos.: All NRC Licensees
______________________________________/

PETITION UNDER 10 C.F.R. §2.206 SEEKING ENFORCEMENT ACTION AGAINST LICENSEES OF THE U.S. NUCLEAR REGULATORY COMMISSION

NOW COMES, Saprodani Associates, ("Petitioners) by and through the undersigned consulting associate, Thomas Saporito, and submits a *"Petition Under 10 C.F.R. §2.206 Seeking Enforcement Action Against Licensees of the U.S. Nuclear Regulatory Commission"*, (Petition). For the reasons stated below, the U.S. Nuclear Regulatory Commission (NRC) should grant the Petition as a matter of law:

NRC HAS JURISDICTION AND AUTHORITY TO GRANT PETITION

The U.S. Nuclear Regulatory Commission (NRC), is the government agency charged by the United States Congress to protect public health and safety and the environment related to the operation of commercial nuclear reactors in the United States of America (USA). Congress charged the NRC with this grave responsibility in creation of the agency through passing the Energy Reorganization Act of 1974, as amended, 42 U.S.C.A. §5851 (ERA). In the instant action, various utility operators in the USA, are collectively and singularly a "licensee" of the NRC and subject to NRC regulations and authority under 10 C.F.R. §50 and under other NRC regulations and authority in the operation of nuclear reactors within the continental United States. Thus, through Congressional action in creation of the agency; and the fact that the named-actionable parties identified immediately above by Petitioners are collectively and singularly a licensee of the NRC, the agency has jurisdiction and authority to grant the Petition.

STANDARD OF REVIEW

A. Criteria for Reviewing Petitions Under 10 C.F.R. §2.206

The staff will review a petition under the requirements of 10 C.F.R. §2.206 if the request meets all of the following criteria:

- The petition contains a request for enforcement-related action such as issuing an order modifying, suspending, or revoking a license, issuing a notice of violation, with or without a proposed civil penalty, etc.

- The facts that constitute the basis for taking the particular action are specified. The petitioner must provide some element of support beyond the bare assertion. The supporting facts must be credible and sufficient to warrant further inquiry.

- There is no NRC proceeding available in which the petitioner is or could be a party and through which petitioner's concerns could be addressed. If there is a proceeding available, for example, if a petitioner raises an issue that he or she has raised or could raise in an ongoing licensing proceeding, the staff will inform the petitioner of the ongoing proceeding and will not treat the request under 10 C.F.R. §2.206.

B. Criteria for Rejecting Petitions Under 10 C.F.R. §2.206

- The incoming correspondence does not ask for an enforcement-related action or fails to provide sufficient facts to support the petition but simply alleges wrongdoing, violations of NRC regulations, or existence of safety concerns. The request cannot be simply a general statement of opposition to nuclear power or a general assertion without supporting facts (e.g., the quality assurance at the facility is inadequate). These assertions will be treated as routine correspondence or as allegations that will be referred for appropriate action in accordance with MD 8.8, "Management of Allegations".

- The petitioner raises issues that have already been the subject of NRC staff review and evaluation either on that facility, other similar facilities, or on a generic basis, for which a resolution has been achieved, the issues have been resolved, and the resolution is applicable to the facility in question. This would include requests to reconsider or reopen a previous enforcement action (including a decision not to initiate an enforcement action) or a director's decision. These requests will not be treated as a 2.206 petition unless they present significant new information.

- The request is to deny a license application or amendment. This type of request should initially be addressed in the context of the relevant licensing action, not under 10 C.F.R. 2.206.

- The request addresses deficiencies within existing NRC rules. This type of request should be addressed as a petition for rulemaking.

See, Volume 8, Licensee Oversight Programs, Review Process for 10 C.F.R. Petitions, Handbook 8.11 Part III.

REQUEST FOR ENFORCEMENT-RELATED ACTION TO MODIFY, SUSPEND, OR REVOKE A LICENSE AND ISSUE A NOTICE OF VIOLATION WITH A PROPOSED CIVIL PENALTY

A. Request for Enforcement-Related Action

Petitioners respectfully request that the NRC take escalated enforcement action against the above-captioned licensee(s) and suspend, or revoke the NRC license(s) granted to the licensee(s) for operation of nuclear power reactors; and that the NRC issue a notice of violation with a proposed civil penalty against the collectively named and each singularly named licensee captioned-above in this matter. In particular, Petitioners request that the NRC ORDER the immediate shut-down of all nuclear power reactors in the USA which are known to be located on or near an earthquake fault-line.

B. Facts That Constitute the Basis for Taking the Requested Enforcement-Related Action Requested by Petitioners

On or about March 11th, 2011, following an "act of GOD" - an 8.9 magnitude earthquake in the country of Japan, one or more nuclear power reactors in Japan sustained significant damage to their Emergency Core Cooling Systems (ECCS) which resulted in the release of radio-active particles from at least one nuclear reactor into the environment in the surrounding areas in Japan. The Japanese authorities ordered a "General Emergency Evacuation"; however, it appears that many Japanese citizens were not able to timely leave the endangered area and are subject to radio-active contamination at this time.

Petitioners aver here that many of the NRC's licensees which operate nuclear power reactors under permissive licenses issued by the NRC under 10 C.F.R. §50, operate said nuclear power reactors on or near earthquake fault lines - and are therefore subject to significant earthquake damage - similar to the recent earthquake damage sustained by the nuclear power reactors witnessed in the country of Japan for which an on-going state of emergency continues to unfold. Moreover, Petitioners aver here that the licensees' safety-analysis and safety design basis relied upon by the NRC in granting operational licenses to the licensee(s) is flawed and will subject said nuclear power reactors to a Loss-of-Coolant-Accident (LOCA) - similar to the LOCA now occurring in the country of Japan. Thus, the immediate actions sought by Petitioners in the instant action on the part of the NRC, are vital in protecting public health and safety in these dire circumstances.

C. There Is No NRC Proceeding Available in Which the Petitioners are or Could be a Party and Through Which Petitioners' Concerns Could be Addressed

Petitioners aver here that there is no NRC proceeding available in which the Petitioners are or could be a party and through which Petitioners' concerns could be addressed.

CONCLUSION

FOR ALL THE ABOVE STATED REASONS, and because Petitioners have amply satisfied all the requirements under 10 C.F.R. §2.206 for consideration of [their] Petition by the NRC Petition Review Board, (PRB), the NRC should grant Petitioners' requests made in the instant Petition as a matter of law.

Respectfully submitted,

Thomas Saporito, Consulting Associate
Saprodani Associates
Post Office Box 8413
Jupiter, Florida 33468-8413
Voice: (561) 972-8363
Email: thomas@saprodani-associates.com

CERTIFICATE OF SERVICE

I HEREBY CERTIFY, that on this 12th day of March, 2011, a copy of foregoing document was provided to those identified below by means shown:

Hon. William Borchardt
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
{Sent via U.S. Mail and electronic mail}

Hon. Gregory B. Jaczko, Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
{Sent via electronic mail}

Carolyn Evans, Dir. of Enforcement
U.S. Nuclear Regulatory Commission
Region II Headquarters
Atlanta, Georgia 30303
{Sent via electronic mail}

Local and National Media Sources

Melanie Checkle, Allegations Coordinator
U.S. Nuclear Regulatory Commission
Region II Headquarters
Atlanta, Georgia 30303
{Sent via electronic mail}

Oscar DeMiranda
Senior Allegations Coordinator
U.S. Nuclear Regulatory Commission
Region II Headquarters
Atlanta, Georgia 30303
{Sent via electronic mail}

By: ______________________
Thomas Saporito

SA20110414.14

NUCLEAR INFORMATION AND RESOURCE SERVICE

6930 Carroll Avenue, Suite 340, Takoma Park, MD 20912
301-270-NIRS (301-270-6477); Fax: 301-270-4291
nirsnet@nirs.org; www.nirs.org

General Electric Mark I Reactors in the United States

The Fukushima Daiichi Unit 1 reactor that exploded on Saturday, March 12, 2011, was a General Electric Mark I reactor. This design has been criticized by nuclear experts and even Nuclear Regulatory Commission staff for decades as being susceptible to explosion and containment failure.

As early as 1972, Dr. Stephen Hanuaer, an Atomic Energy Commission safety official, recommended that the pressure suppression system be discontinued and any further designs not be accepted for construction permits. Shortly thereafter, three General Electric nuclear engineers publicly resigned their prestigious positions citing dangerous shortcomings in the GE design.

An NRC analysis of the potential failure of the Mark I under accident conditions concluded in a 1985 report that Mark I failure within the first few hours following core melt would appear rather likely. In 1986, Harold Denton, then the NRC's top safety official, told an industry trade group that the "Mark I containment, especially being smaller with lower design pressure, in spite of the suppression pool, if you look at the WASH 1400 safety study, you'll find something like a 90% probability of that containment failing."

For more information, see: http://www.nirs.org/factsheets/bwrfact.htm

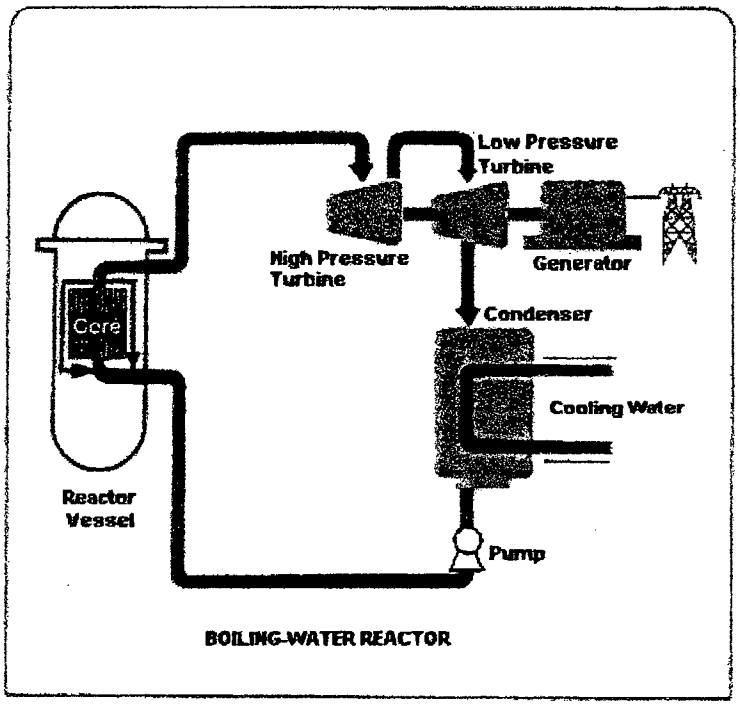
Reactor	Location	Size	Year operation began
Browns Ferry 1*	Decatur, AL	1065 MW	1974
Browns Ferry 2*	Decatur, AL	1118 MW	1974
Browns Ferry 3*	Decatur, AL	1114 MW	1976
Brunswick 1*	Southport, NC	938 MW	1976
Brunswick 2*	Southport, NC	900 MW	1974
Cooper*	Nebraska City, NE	760 MW	1974
Dresden 2*	Morris, IL	867 MW	1971
Dresden 3*	Morris, IL	867 MW	1971
Duane Arnold*	Cedar Rapids, IA	581 MW	1974
Hatch 1*	Baxley, GA	876 MW	1974
Hatch 2*	Baxley, GA	883 MW	1978
Fermi 2	Monroe, MI	1122 MW	1985
Hope Creek**	Hancocks Bridge, NJ	1061 MW	1986
Fitzpatrick*	Oswego, NY	852 MW	1974
Monticello*	Monticello, MN	572 MW	1971
Nine Mile Point 1*	Oswego, NY	621 MW	1974
Oyster Creek*	Toms River, NJ	619 MW	1971
Peach Bottom 2*	Lancaster, PA	1112 MW	1973
Peach Bottom 3*	Lancaster, PA	1112 MW	1974
Pilgrim**	Plymouth, MA	685 MW	1972
Quad Cities 1*	Cordova, IL	867 MW	1972
Quad Cities 2*	Cordova, IL	867 MW	1972
Vermont Yankee*	Vernon, VT	620 MW	1973

*has received 20-year license extension from the Nuclear Regulatory Commission
**20-year license renewal extension is under review by Nuclear Regulatory Commission

HAZARDS OF BOILING WATER REACTORS IN THE UNITED STATES

BACKGROUND

Of the 104 operational nuclear power reactors in the United States, thirty-five are boiling water reactors (BWR). General Electric is the sole designer and manufacturer of BWRs in the United States. The BWR's distinguishing feature is that the reactor vessel serves as the boiler for the nuclear steam supply system. The steam is generated in the reactor vessel by the controlled fissioning of enriched uranium fuel which passes directly to the turbogenerator to generate electricity.



LACK OF CONTAINMENT INTEGRITY DURING A NUCLEAR ACCIDENT

The purpose of a reactor containment system is to create a barrier against the release of radioactivity generated during nuclear power operations from certain "design basis" accidents, such as increased pressure from a single pipe break. It is important to understand that nuclear power plants are not required by the Nuclear Regulatory Commission (NRC) to remain intact as a barrier to all possible accidents or "non-design basis" accidents, such as the melting of reactor fuel. All nuclear reactors can have accidents which can exceed the design basis of their containment.

But even basic questions about the the GE containment design remain unanswered and its integrity in serious doubt. For example, 23 of these BWRs use a smaller GE Mark I pressure suppression containment conceived as a cost-saving alternative to the larger reinforced concrete containments marketed by competitors. A large inverted light-bulb-shaped steel structure called "the drywell" is constructed of a steel liner and a concrete drywell shield wall enclosing the reactor vessel--this is considered the "primary" containment.. The atmosphere

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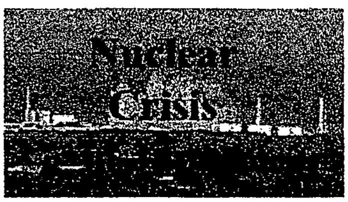


of the drywell is connected through large diameter pipes to a large hollow doughnut-shaped pressure suppression pool called "the torus", or wetwell, which is half-filled with water. In the event of a loss-of-coolant-accident (LOCA), steam would be released into the drywell and directed underwater in the torus where it is supposed to condense, thus suppressing a pressure buildup in the containment.

The outer concrete building is the "secondary" containment and is smaller and less robust (and thus cheaper to build) than the containment buildings used at most reactors.

As early as 1972, Dr. Stephen Hanauer, an Atomic Energy Commission safety official, recommended that the pressure suppression system be discontinued and any further designs not be accepted for construction permits. Hanauer's boss, Joseph Hendrie (later an NRC Commissioner) essentially agreed with Hanauer, but denied the recommendation on the grounds that it could end the nuclear power industry in the U.S.

Here are copies of the three original AEC memos, including Hendrie's:

November 11, 1971: outlines problems with the design and pressure suppression system containment.

September 20, 1971: memo from Steven Hanauer recommends that U.S. stop licensing reactors using pressure suppression system

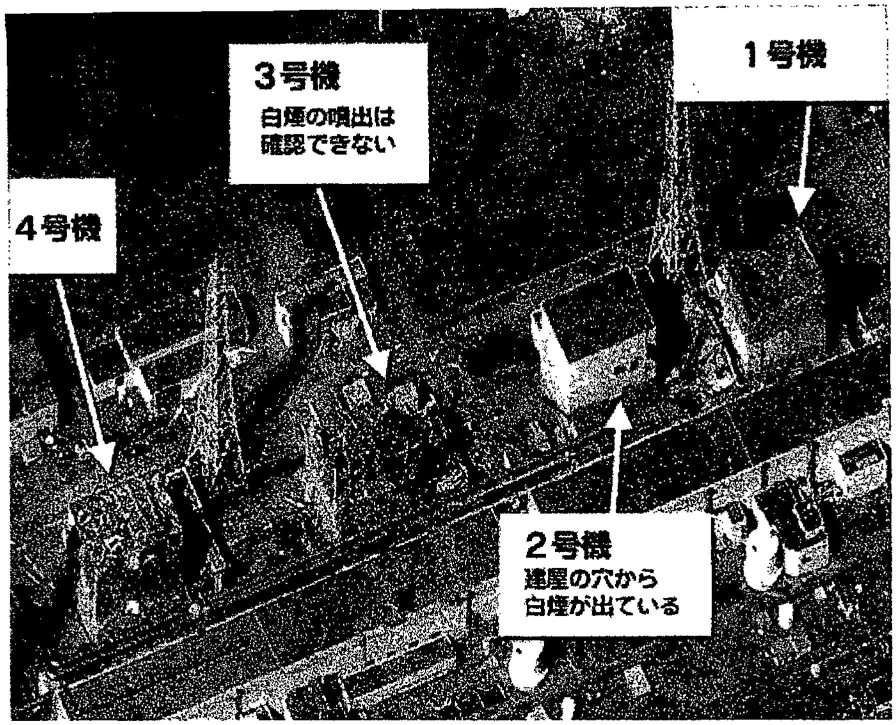
September 25, 1972: memo from Joseph Hendrie (top safety official at AEC) agrees with recommendation but rejects it saying it "could well mean the end of nuclear power..."

In 1976, three General Electric nuclear engineers publicly resigned their prestigious positions citing dangerous shortcomings in the GE design.

An NRC analysis of the potential failure of the Mark I under accident conditions concluded in a 1985 report that Mark I failure within the first few hours following core melt would appear rather likely."

In 1986, Harold Denton, then the NRC's top safety official, told an industry trade group that the "Mark I containment, especially being smaller with lower design pressure, in spite of the suppression pool, if you look at the WASH 1400 safety study, you'll find something like a 90% probability of that containment failing." In order to protect the Mark I containment from a total rupture it was determined necessary to vent any high pressure buildup. As a result, an industry workgroup designed and installed the "direct torus vent system" at all Mark I reactors. Operated from the control room, the vent is a reinforced pipe installed in the torus and designed to release radioactive high pressure steam generated in a severe accident by allowing the unfiltered release directly to the atmosphere through the 300 foot vent stack. Reactor operators now have the option by direct action to expose the public and the environment to unknown amounts of harmful radiation in order to "save containment." As a result of GE's design deficiency, the original idea for a passive containment system has been dangerously compromised and given over to human control with all its associated risks of error and technical failure.

As we have now seen at Fukushima, Japan, in March 2011, this containment design failed catastrophically when hydrogen built up in the outer containment buildings until three of them exploded. The outer containment building was neither large enough nor strong enough to withstand these explosions.



VULNERABILITY OF IRRADIATED FUEL POOLS

The irradiated (sometimes called "spent") fuel pools in GE Mark I reactors are above the reactor core and outside the primary containment system. This design was chosen for efficiency, not safety--the fuel rods in the reactor are lifted by crane and simply moved over to the fuel pool. The explosions at Fukushima that caused severe damage to the containment buildings (as can be seen in the above satellite photo taken March 18, 2011) also exposed and compromised the fuel pools providing a direct pathway for release of radioactivity into the air. While there was substantial amounts of fuel in the Fukushima pools, in the U.S. pools are typically packed even more densely, meaning even higher potential radiation risks if they are compromised.

DETERIORATION OF BWR SYSTEMS AND COMPONENTS

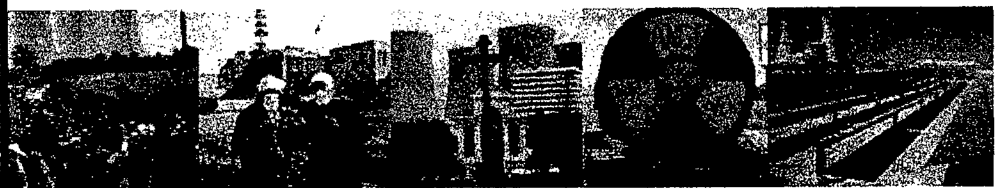
It is becoming increasingly clear that the aging of reactor components poses serious economic and safety risks at BWRs. A report by NRC published in 1993 confirmed that age-related degradation in BWRs will damage or destroy many vital safety-related components inside the reactor vessel before the forty year license expires. The NRC report states "Failure of internals could create conditions that may challenge the integrity the reactor primary containment systems." The study looked at major components in the reactor vessel and found that safety-related parts were vulnerable to failure as the result of the deterioration of susceptible materials (Type 304 stainless steel) due to chronic radiation exposure, heat, fatigue, and corrosive chemistry. One such safety-related component is the core shroud and it is also an indicator of cracking in other vital components through the reactor made of the same material.

Core Shroud Cracking

The core shroud is a large stainless steel cylinder of circumferentially welded plates surrounding the reactor fuel core. The shroud provides for the core geometry of the fuel bundles. It is integral to providing a refloodable compartment in the event of a loss-of-coolant-accident. Extensive cracking of circumferential welds on the core shroud has been discovered in a growing number of U.S. and foreign BWRs. A lateral shift along circumferential cracks at the welds by as little as 1/8 inch can result in the misalignment of the fuel and the inability to insert the control rods coupled with loss of fuel core cooling capability. This scenario can result in a core melt accident. A German utility operating a GE BWR where extensive core shroud cracking was identified estimated the cost of replacement at $65 million dollars. The Wuergassen reactor, Germany's oldest boiling water reactor, was closed in 1995 after wary German nuclear regulators rejected a plan to repair rather than replace the reactor's cracked core shroud.

Rather than address the central issue of age related deterioration, U.S. BWR operators now opt for a dangerous piecemeal approach of patching cracking parts at least cost but increased risk.

Nuclear Information and Resource Service



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It's time: demand permanent shutdown of GE Mark I reactors!

There are 23 nuclear reactors currently operating in the United States using the same General Electric Mark I design as those that have failed so catastrophically at Fukushima.

The flaws in this design are fundamental, cannot be fixed, and have been documented for 40 years now. They have led to containment building explosions at three of the Fukushima reactors, the exposure of irradiated fuel pools to the environment and enormous radiation releases.

It is time to close them, permanently. Please take a moment to tell that to President Obama, your Congressmembers, and the Nuclear Regulatory Commission (NRC). You can do that with one letter below.

A list of the GE Mark I reactors can be found here. You will also see that 18 of these 23 aging reactors already have been approved for license extension by the NRC. Rather than putting Americans into peril for 20 more years, the renewals must be rescinded and the reactors closed now.

An explanation of the fundamental flaws in the Mark I design is here. The explanation includes links to the original 1971-72 Atomic Energy Commission discussion of the design's flaws and recommendation that the design be discontinued in the U.S. Astonishingly, this recommendation was agreed to in concept, but denied because it "could well mean the end of nuclear power..."

Please use the icons above to spread the word about this campaign via e-mail and social networking sites! Your help in outreach is crucial.

People outside U.S.: Because this campaign is partially aimed at U.S. Congressmembers, only U.S. addresses may be used. We ask you instead to use the letter below, edit as you wish, and e-mail it to President Obama from http://www.whitehouse.gov/contact.

Dear President Obama,

There are 23 operating and aging GE Mark I reactors in the United States. This is the same design that has failed so catastrophically at Fukushima.

Top safety officials at the Atomic Energy Commission and later the Nuclear Regulatory Commission have warned about the flaws of this reactor design for the past 40 years.

The flaws in this design are fundamental and cannot be fixed.

Americans should not live in peril due to flawed reactor designs. Taken together, all 23 of these reactors provide less than 4% of the nation's electricity. There is ample reserve capacity available.

These reactors must be permanently closed now. Please inform me of the actions you will take to ensure their permanent shutdown.

Subject:

Close All GE Mark I Reactors Now

Your Letter: