OPSMPEm Resource

From: West, Stephanie
Sent: Monday, March 11, 2013 9:46 AM
To: OPSMNPEm Resource; OPSMPEm Resource
Subject: Blog posts for January and February 2013
Attachments: blog-published-2013-03-05@14-51-38.pdf

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Happy New Year and Some NRC Blog Updates

posted on Thu, 03 Jan 2013 14:41:59 +0000

Eliot Brenner
Public Affairs Director

Two years ago, at the end of this month, the NRC made its first foray into social media with this blog. We didn't know what to expect, but we knew we needed a new way to provide information to the public about – and explanations of – the important, but frequently complex and technical activities of the agency. We were pleased with the initial interest in the blog, which sky-rocketed during the March 2011 events in Japan, and which spiked again during the August 2011 earthquake in Virginia and this fall's Hurricane Sandy. Some stats for 2012: we put up 138 posts and approved 893 comments, and got some 126,000 views. And we have 700+ subscribers. Also during those 12 months, the NRC sent out 540 tweets (to 2,996 followers), posted 47 videos on YouTube (which got 17,170 views) and posted 1,100 photos on Flickr, with 93,000 views. We are continuing to look at new ways to make our social media program interesting and relevant to you, and we may have some news in that regard later in the year. So stay tuned. We are making a few small changes to the blog for 2013. We'll be using more specific tags to make it easier to find our content. We're also moving the bylines up to the top, so you'll know right away who has authored the post. And when we can, we'll be adding more links and more interactive content. We hope to continue to see the lively conversations in the comments to our posts. We approve and post comments several times a day to reduce delaying the conversation. In only a few instances, we have not approved comments because they didn't meet our blog comment guidelines. If the comment is determined to be an allegation, for example, it will not be posted (but you will be contacted by our allegations team.) If your comment is a personal attack, we also cannot post it. Some comments lately have drifted into this category, so we ask that you be respectful to other commenters and their points of view so that we can approve your comments. Also remember that comments need to be related to the post under which they are submitted. If they're not, we'll move them to the Open Forum section. We encourage you to post there when there's something you want to talk about, but about which there is no recent post. Which brings me to my last point. We would love to know what you'd like us to cover for the new year. What topics are of interest to you? What do you want to know more about – or perhaps get a plain language explanation of? Please let us know in the comment section below and we'll try to tailor future content to your needs. Happy New Year to you, and thank you for reading the NRC's blog.

Comments

comment #39620 posted on 2013-01-03 14:37:21 by fresh

> I would like to see a realistic discussion of the scenario of another Carrington Event that fries 80% of all the big transformers in a multi-state zone, and what happens after 2 weeks of oil for the generators runs out and cannot be supplied at any cost.

comment #39633 posted on 2013-01-03 15:34:39 by Joffan in response to comment #39571

> ... and comment preview would nice... as you can tell from my mistakes above...

comment #39571 posted on 2013-01-03 11:52:59 by Joffan

> Congratulations on another year of blogging. Some interesting posts and insights in NRC thinking. I wasn't aware of your open forum. There are a couple of things about it that could be improved: 1. Comments moved appear to all be assigned to "Moderator"; it would be better if the original commenter was still shown for the post, even if 90% are CaptD rambling on about conspiracies at San Onofre. 2. There are some many comments that a comment page number access is required, rather than simply "Older" and "Newer" navigation. 3. Some of the moved posts have lost context to the extent that they are meaningless. I'm not sure it's the best way to capture ongoing discussion, since there is little opportunity to organize into discussion threads, but I do appreciate Possible future

topics: - Regulation of hospitals - Food irradiation - Yucca Mountain safety case

comment #39574 posted on 2013-01-03 12:02:55 by Garry Morgan

Thumbs up for your social media interaction!

comment #40570 posted on 2013-01-07 22:37:32 by James Greenidge in response to comment #40186

Ditto, KR. If "fresh"'s maximum Carrington frets ever came true, nuclear plants -- which in all exampled likelihood wouldn't harm anyone even in unattended full meltdown -- would literally be the last of my concerns what the social state the country would be in. Besides I trust that our engineers and techs are bright enough to've foreseen the issue and solutions even before fresh has! James Greenidge Queens NY

comment #40575 posted on 2013-01-08 01:19:11 by HelpAllHurtNeverBaba

Mr. Eliot Brenner, Happy New Year to You and All the NRC Staff and their families San Onofre Unit 2 Steam Generators are not Safe at any Power due to fluid elastic instability and do not meet the NRC Licensed Conditions for Operation to RTP and Design Basis Accident Conditions. Fluid Elastic Instability is a very controversial and complex phenomenon, which has been around since 1966. To understand the concept of FEI in a two-phase heterogeneous steam water mixture in a Nuclear Steam Generators such as SONGS, Researchers and Steam Generator Designers/Manufacturers are still struggling. FEI can be caused by a combination of low steam generator pressures, low steam saturation temperatures, low tube clearances, high steam flows, high fluid velocities, improperly designed floating anti-vibration structures, no-in plane supports, potential of corrosion products clogging the TSP holes, lack of mixing baffles in the steam generator lower section, narrow tube pitch to tube diameter, low tube wall thickness to diameter ratio, high tube length to tube diameter ratio, tube frequency, resonance vibrations, reactor coolant pump pressure pulses and several other unknown factors. The bottom line of FEI is the absence of water film on the steam generator tubes AKA localized steam voids, Zero Damping, etc.. SONGS MHI RSGs suffer from most of these problems. On top of these problems, these “defectively designed and degraded generators” have thousands of damaged and cracked tubes. Therefore, these generators cannot meet the 10 CFR Part 50 Appendix A, General Design Criteria 14 and SONGS Unit 2 Technical Specifications structural integrity performance criterion in TS 5.5.2.11.b.1. APS/Westinghouse/Combustion Engineering solved all these problems in Palo Verde Nuclear Generating Station Replacement Steam Generators via a 50.90 License Amendment Process and thorough scrutiny by NRR Brilliant Engineers and Experts from other NRC Divisions. When I see examples of work like this, I am proud of NRC but SONGS is all messed up. NRC Region IV really needs to do a self-assessment to figure out what went wrong at SONGs and not come up with excuses because of the careless attitude of a few. Just my humble suggestion. If SCE, NRC Region IV and CPUC want safe, dependable and profitable power and win public confidence, SCE needs to return $1 Billion to Rate Payers and pay another 1 Billion Dollars to Westinghouse (The Most Advanced, Sophisticated, Skilled and Dependable US Nuclear Plant Designer & Manufacturer) to remove the defective generators and build new replacement generators designed to produce licensed safe power like Palo Verde Nuclear Generating Station. MHI & SCE cannot fix these Defective and Degraded replacement steam generator. Thanks...

comment #40186 posted on 2013-01-06 00:04:50 by KR in response to comment #39620

Because that is realistic? If we seriously couldn't get diesel fuel at any cost, there would be a lot more problems than the local nuclear power plant. And I'm pretty sure the military will have already stepped up and provided or moved that fuel.

NRC Public Meetings – Deciphering the Categories

posted on Tue, 08 Jan 2013 16:43:26 +0000

[caption id="attachment_3575" align="alignright" width="300"] A public meeting about the San Onofre nuclear power plant draws a large crowd.[/caption]
Lance Rakovan
Senior Communications Specialist

One of my jobs at the NRC is managing the agency’s Meeting Facilitation and Advisor Program. This means, I train employees to facilitate public meetings and recommend ways to make public meetings as meaningful as possible. One of the challenges I face is explaining – both to those inside and outside the agency – what the three meeting categories mean. The answer? Simply this: the category of the meeting is a reflection of the purpose of the meeting.Category 1 meetings, for example, are between the NRC and one other party – typically a licensee of the NRC, a vendor, or an applicant or potential applicant for a license. The NRC has these types of meetings in a public forum to provide transparency even though the purpose is to have a one-on-one discussion. The public can observe the meeting and has the opportunity to ask questions of the NRC after the business portion of the meeting, but doesn't participate in the discussion itself. Category 2 meetings are between the NRC and a number of individuals representing groups such as licensees, vendors, other federal agencies, or non-governmental organizations. Like Category 1 meetings, the NRC holds these meetings in a public forum. The purpose of the meeting is for the NRC to conduct a discussion with the designated group. The public can observe the meeting and ask questions of the NRC after the business portion of the meeting, but again, doesn't participate in the discussion itself. A common type of Category 2 meeting is a roundtable meeting where the NRC invites representatives of the broad spectrum of interests affected by an issue to engage in discussion with each other and the NRC, with the public in an observing role. Category 3 meetings are fully engaged discussions between the NRC and the public (as well as stakeholders that might include other government agencies, the industry and others). Public participation is actively sought at this type of meeting, which has the widest participation opportunities and is specifically tailored for the public to comment or ask questions. Category 3 meetings are also known as Town Hall meetings. We might hold such a meeting to inform the public about a particular issue, respond to questions or receive comments from attendees. These meetings might be preceded by other information opportunities, such as a poster session or open house. We do our best to conduct public meetings that not only accomplish a particular purpose, but also allow the public to observe and participate. The meeting category just gives an indication of what kind of meeting you can expect. Additional Information on our public meeting policies can be found on our web site.

Comments

comment #42468 posted on 2013-01-15 00:15:43 by James Greenidge in response to comment #40964

Re: "I agree that a facilitator should not become the focus of a meeting, however I believe one of the facilitator's duties is to make sure that meeting participants understand each other." Lance, I both admire, sympathize and don't envy the NRC crews at these meetings, encountering and trying to educate many whom are simply philosophically implacable at accepting any reassurance of proof or fact or enlightenment. My hat's off to all of you! James Greenidge Queens NY

comment #40909 posted on 2013-01-09 08:33:48 by in response to comment #40722

Completely agree with you CaptD. This filibustering technique is an effective way to "chill" the audience and unfortunately it is practiced too often in DC.

comment #40964 posted on 2013-01-09 12:46:13 by Moderator in response to comment #40722

I appreciate your comments on our public meetings and facilitation practices and would like to weigh in on a few of the points you made. I agree that a facilitator should not become the focus of a meeting, however I believe one of the facilitator's duties is to make sure that meeting participants understand each other. When necessary, a facilitator should step in to re-phrase a comment if he or she believes the comment was unclear somehow to other participants. Hopefully, this can be done quickly as to avoid taking too much time away from public participation. I am happy to communicate with NRC facilitators again the need for them to work towards having a minimal presence at a meeting and ensuring that all participants have an opportunity to speak. We are always working towards improving upon our skills, and striving to make meetings as productive as possible. I hope that you will continue to participate in our public meetings and that you will continue to provide input on how we can make improvements. Lance Rakovan

comment #41006 posted on 2013-01-09 16:32:58 by HelpAllHurtNeverBaba

NRC Public Category 3 Meetings are good provided public is given a chance to ask questions. In the last SCE NRC Public Meeting on November 30, 2012, I raised my hand for more than 30 minutes to give vital information to the NRC Panel, but the Moderator Turned away always giving me a big smile and denying a chance to ask a question by handing microphone to somebody else. The question is why....Thanks

comment #40778 posted on 2013-01-08 15:08:55 by James Greenidge

This is a noble and admirable but often thankless effort at public education, considering that most the public attending walk in with bogeymen of FUD saddled their backs. If there was a way to also level the playing field and soothe blind passions for rational discussion by a initial short film/lecture on how nuclear energy operations work and nuclear's near impeccable safety record for 60 years around the world and such, it'd do greatly in assuaging any Doomsday jitters that it's all over if a hammer just drops anywhere inside the giant eggshells of nuclear plants! Wish I was there! Keep chugging and FUD busting! James Greenidge Queens NY

comment #40722 posted on 2013-01-08 12:02:33 by CaptD

Lance, You need to do a much better job training your Facilitators, A Facilitator should not become the focus of the meeting and or "re-explain" what previous comments mean, but rather insure that the audience gets the maximum amount of time to interact with the NRC on the rare times when these meetings are open to actual public participation! The last SCE - NRR meeting is a perfect example of what not to do, the Facilitator was trying to take the limelight and several in the audience even called him on it! Further those waiting to speak via the phone were angered when he said that no one else was waiting to speak, when I know of at least two people that had been on hold to speak after having been questioned by whoever was running that part of the show! The NRC needs to step up and change the way it handles meetings if it is going to fulfill its "NRC Public Meetings and Involvement", which states, "The NRC considers public involvement in, and information about, our activities to be a cornerstone of strong, fair regulation of the nuclear industry." BTW: The meetings videos need to be improved as the quality is often poor, no pun intended!

comment #40720 posted on 2013-01-08 11:56:10 by richard123456columbia

Is there a way for the public to ask for a topic to be discussed or are the meeting topics only the owners and supporters of nuclear power. If there has been would you please send SITE info for my eyes, thanks.

Force-on-Force or Was That a Gunfight at a Nuclear Power Plant?

posted on Tue, 15 Jan 2013 13:49:48 +0000

Clay Johnson
Chief, Security Performance Evaluation Branch

They are dressed in camouflage, fit and well-trained, and they creep quietly toward the perimeter of a nuclear power plant under cover of darkness. Their realistic weapons reflect dully in the moonlight, but these weapons fire blank ammunition and lasers that record hits and misses. Their goal? A particular target set within the plant which, if compromised, could impact the safety of the plant and the community that surrounds it. The target set this night? A closely guarded secret known only to the "armed intruders" and the NRC inspection team that includes active duty military members from the U.S. Special Operations Command. The attacks will be repeated over the course of three days and nights so that different attack methods and various targets at each nuclear power plant are tested. In each scenario, the plant's security personnel work to protect specific areas of the plant according to their facility's individual security plan. Each plant is tested in this manner every three years. These force-on-force inspections have been part of the NRC inspection regime since 1991, but they were significantly beefed up and the frequency increased to every three years after Sept. 11, 2001. They are designed to assess the plant's ability to defend itself against the conditions put forth under the "design basis threat" or DBT. These inspections are in addition to the baseline security inspections performed by the NRC's regional inspectors and the inspections done daily by the NRC's resident inspectors. NRC security experts routinely review options for further enhancements to the program. The details of what happens during a force-on-force inspection are not public due to the sensitive nature of security plans at the plants. If a deficiency is found during an inspection, the NRC inspectors stay on site until compensatory measures are put in place, and then the NRC reviews the plant's long-term plan to rectify the problem, and may issue violations. These violations are only discussed in a general way with the public. The "bad guys" are part of what is called the Composite Adversary Force and they are contracted by the nuclear industry to perform these mock attacks to NRC specifications. The plant knows the force-on-force will occur at a specific date for safety and logistical purposes and to provide time to coordinate two sets of security offices – one to participate in the inspection and one to maintain the security posture of the plant. The mock attacks are also preceded by significant planning and on-site tabletop drills conducted by the NRC inspection team. These realistic and physically intensive exercises are but one vehicle by which the NRC ensures the country's nuclear power plants and Category I fuel facilities are prepared and able to protect themselves. Meetings on possible additional enhancements to this inspection program will be announced in the future.

Comments

comment #42589 posted on 2013-01-15 10:31:38 by Garry Morgan

Security contractors bottom line is money, not security. Repetitive failures relating to nuclear facility security where contractors are involved are inexcusable and unacceptable. Special physical security military forces and DOD police are the only security which should be utilized at ALL nuclear facilities. Security goes further than force on force contractor attacks following a specified contractor attack plan. This type of evaluation is not inclusive of all threats and indicative of a real world multi-front threat and attack scenario. Three year security test cycles are inadequate; force on force tests should occur annually. Broad scale systems security tests should also involve defenses to terrorism, cyber attacks, threats to the civilian community, engineering weaknesses, personnel reliability weaknesses, aerial attacks and various diversion and deceit tactics to capture or destroy a facility in various scenarios. The attack scenario evaluation should involve a mass attack of all systems and an extended multi-front attack scenario to include guerrilla Special Forces attacks. Intentional public deceit by corporations, contractors, employees and the regulator is another security risk and threat not addressed.. Intentionally deceiving the public destroys credibility. Nuclear power plant security issues are discussed in this Congressional Research Service (CRS) Report: http://www.fas.org/sgp/crs/homesec/RL34331.pdf What is revealing, the deficiencies noted in a 2008 CRS report still exist in the 2012 report. In the nuclear power industry, money has been placed before the safety and security of citizens.

comment #43330 posted on 2013-01-17 12:08:40 by Garry Morgan in response to comment #42589

Thank you for posting my comment. Security of facilities is vitally important

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A Fire at South Texas Project – How the NRC Responded

posted on Thu, 10 Jan 2013 13:24:16 +0000

Victor Dricks
Senior Public Affairs Officer
Region IV

[caption id="attachment_3601" align="alignright" width="300"]

The Region IV Incident Response Center during an emergency exercise last month.[/caption] At 4:40 p.m. Central Time Tuesday, officials at the South Texas Project nuclear power plant near Bay City, Texas, notified the NRC's Operations Center that a fire had broken out in the main transformer of Unit 2, causing an automatic shutdown. Unit 1 was unaffected and continued to operate at full power. As designed, the plant's emergency diesel generators energized to power safety-related equipment. All four auxiliary feedwater pumps started as required to supply power to the plant's steam generators for cooling. However, power to non-safety related electrical buses was lost, cutting off power to the plant's reactor coolant pumps. As part of the plant's design, natural draft circulation continued to cool the plant's shutdown reactor to remove decay heat. The plant declared an Unusual Event – the lowest of four categories of nuclear emergency -- due to the transformer fire at 4:55 p.m. The plant's on-site fire brigade responded and quickly extinguished the blaze, so no off-site assistance was required. The NRC's resident inspector, who was on-site at the time, responded to the event by going to the plant's control room to observe the licensee's response to the event. The NRC's Region IV Office in Arlington, Texas, activated its Incident Response Center to monitor the event. There were no personnel injuries and no radiological releases were reported. The Unusual event was terminated at 7:47 p.m., although the NRC's resident inspector remained onsite until about midnight. As part of its ongoing oversight, the NRC will monitor the licensee's follow-up actions. These include identification of the cause of the transformer fire; a review of the behavior of the plant's electrical protection systems; and various repair activities. "Overall, from what we now know, plant operators responded well to the event," said Acting Deputy Regional Administrator Steve Reynolds. "The NRC will conduct an independent and comprehensive assessment of this incident as part of its oversight process."

Comments

comment #45362 posted on 2013-01-21 19:41:19 by Hiddencamper in response to comment #43358

> There are also a handful of plants (Byron, Braidwood, Grand Gulf) who have diesel engine driven aux feed pumps. Also, regardless of whether or not they were motor driven aux feed pumps, they would have been on the safeguards bus if AC power was lost. Either way, I'm a big fan of diversity in aux-feed systems, and wish more plants had a combination of all three engine, steam, and motor driven aux feed.

comment #41494 posted on 2013-01-10 20:20:59 by James Greenidge

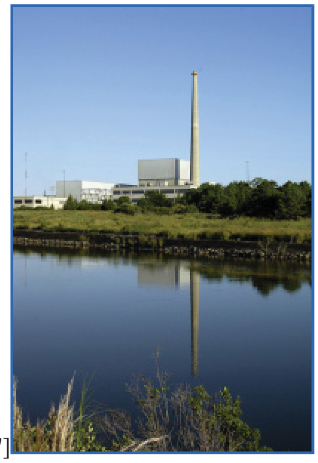
> Interesting memo! Of course, in order to give calm perspective for the public as to the level of the incident it'd really help immensely for the NRC to equate it with other industrial and energy/chemical refinery incident comparisons. Unfortunately, largely due political factors, such facilities aren't as required to issue such reports, despite a far more lethal and damaging track record. James Greenidge Queens NY

comment #43358 posted on 2013-01-17 13:16:50 by praengineer

> This is what gives folks in the industry pause: "All four auxiliary feedwater pumps started as required to supply power to the plant's steam generators for cooling." As most know, the auxiliary feedwater pumps send water to the steam generators. Cooling is a result of boiling off the water provided from auxiliary feedwater. At many plants, auxiliary feedwater pumps are connected to small steam turbines so only rely on DC power for control. The summary in this blog ought to have mentioned whether or not all four were electric motor driven pumps. Only electric motor driven auxiliary feedwater pumps depend on AC power in the plant.

NRC Reports on Oyster Creek Hurricane Performance

posted on Fri, 11 Jan 2013 19:18:04 +0000

[caption id="attachment_3609" align="alignright" width="201"] A photo of the Oyster Creek Nuclear Generating Station, Unit 1, located near Forked River, N.J.
Courtesy:©Exelon Nuclear[/caption] *Neil Sheehan Public Affairs Officer Region I* The NRC staff has issued the findings of the Special Inspection it conducted at the Oyster Creek nuclear power plant to review events related to "Superstorm" Sandy. The inspection was launched on Nov. 13. Our three-member team's primary focus was the timing of the emergency declarations at the Lacey Township (Ocean County), N.J. facility during the storm. Sandy-generated high water levels at the plant's water intake structure, prompting first an "Unusual Event" declaration and later an "Alert" declaration. The inspectors also reviewed preparations by Exelon, the plant's owner, prior to the storm's arrival; equipment performance; and overall command and control from an emergency preparedness perspective. The inspectors' report is now available on the NRC website. The team has concluded that the declarations were timely and accurate and that plant personnel appropriately carried out their duties during the storm. At the same time, the inspectors did observe several areas where performance could be improved. Some examples included heightened awareness of emergency declaration thresholds, clearer documentation in control room records and ensuring reliable back-up power for the plant's emergency operations facility. The report also contains a company-identified violation determined to be of very low safety significance related to the use of incorrect meteorological tower data. In general, the report underscores how plant operators dealt with the harsh conditions at the water intake structure and other challenges, such as the loss of off-site power for a time. While the Special Inspection is finished, the NRC's Resident Inspectors at Oyster Creek will provide additional observations about plant performance during the storm in an upcoming report. What's more, an NRC Petition Review Board continues working on a petition, submitted by several environmental organizations, that raises questions regarding plant performance during the storm. The board on Jan. 3rd conducted a public meeting with the petitioners to gather more information about their concerns. As NRC staff made clear, their goal was to listen to the petitioners, though the staff did explain why the NRC denied the petitioners' request to keep Oyster Creek shut down following the storm.

Comments

comment #42603 posted on 2013-01-15 11:30:58 by LillyMunster in response to comment #42591

> My concern is that they figured all of this out in the middle of a storm and it was apparently missed in all prep over the years. That the actual pump height was 10 feet not 7.4 and that they had no measuring capability over those levels. Having the measuring capability only to the NRC alert level satisfies notifying the NRC at the right level but does nothing for actually dealing with rising water vs. the pump motors so they could de-power them before water would hit.

comment #42089 posted on 2013-01-13 09:56:01 by Nuclear guy

> The good thing is the report is Very consistent with all reports I've seen from legitimate sources during the hurricane. The sad thing is there are likely going to be people who are disappointed that nothing wrong was found.

comment #42109 posted on 2013-01-13 12:02:49 by Nancy in response to comment #41825

> Complimenting the NRC is a bit strange seeing Oyster Creek SFP would have boiled in 28 hours if operator action had not been taken. It was a close call and action is needed to avoid several safety problems learned from this incident.

comment #42591 posted on 2013-01-15 10:32:46 by Moderator in response to comment #41680

> The installed level measuring equipment is adequate for detecting and transmitting levels which translate to emergency action level declarations (greater than 4.5 feet above mean sea level for an "Unusual Event" and greater than 6 feet above mean sea level for an "Alert") and those that procedurally require a change to plant operations (greater than 4.5 feet above mean sea level – commence a normal plant shutdown; and greater than 6 feet above mean sea level – scram the reactor). Once above the 6-foot level, the only actions that can be taken are to shut off electric pumps before their motors are flooded. As a corrective action following "Superstorm" Sandy, Oyster Creek is installing a visible mark, below the bottom of the motor on each of the pumps which could be affected by a high intake level. The mark gives operators an easily viewed indication to tell them when to shut a particular pump off. If intake level ever reached the point at which the pump motors were submerged, there are no other actions for operators to take, so accurate measurements of the level would not a determinant on how to proceed next. Neil Sheehan

comment #42611 posted on 2013-01-15 11:49:28 by LillyMunster in response to comment #42109

Hiddencamper tries to blind people with a wall of poorly organized techno-babble that really clouds the actual areas of concern. 1. "feed and bleed" ie: dumping water into the spent fuel pool is NOT cooling. The loss of the intake pumps loses spent fuel pool cooling capability. The replacing water into an evaporating pool creates lots of other challenges like a refueling floor filling full of steam and excess moisture. If the SGTS isn't available there isn't a way to pull steam out of the refueling floor. Loss of AC power would disable the SGTS so if the generator dropped they have yet another problem and the multiple problems or cascading problems are when the bigger trouble begins. This is why people have so much concern, too many times these situations are highly dependent on hoping one piece of equipment keeps working so things don't start spiraling out of control. Oyster Creek does not have a back up spent fuel pool cooling system and needs one. The NRC needs to make that upgrade get done sooner, not later. Sandy points out exactly why that improvement needs to be done. 2. Portable pumps make no difference if there is no way to operate the spent fuel pool cooling loop. Since they have no back up power system yet if they lose AC power they lose that system. I am also not aware of any actual changes done to tie in a portable pump to the spent fuel pool cooling loop to operate it using a portable pump to move the sea water side of the loop. Again, the public is largely left in the dark. 3. Fire water injection is an emergency measure. Good luck getting a fire truck in if a hurricane trashes the one access bridge into Oyster Creek. If your down to injecting water with a fire truck there are bigger issues going on. 4. The 22 foot flood level at Oyster Creek is utterly irrelevant to the actual public concerns with what went on at OC. Since the intakes have to shut down at 7.4 to 10 feet and then loses plant access to the ultimate heat sink it puts the plant into a challenged condition and the public has every right to not be amused by such a situation especially during a hurricane where most of the warning sirens were down and the ability to evacuate would have been impossible. 5. Loss of AC power is a legit concern. My biggest concern with this incident is the lack of decent communication. The public lacks access to a decent overview of plant systems. This caused lots of information gaps that actually caused MORE public concern than if the public had that information. The conflicting and days later information made it worse. Don't be indignant that the public is concerned about a situation at a nuclear plant when the operator refuses to be accountable to the public and the NRC only informs the public of critical information months later.

comment #42904 posted on 2013-01-16 10:52:04 by LillyMunster in response to comment #42712

Jeez. it is "fact" because you say so? Dumping water into the pool is not true cooling even if it has the function of helping to cool the temp. This is the problem. There are so many so busy trying to defend things by redefining reality. HVAC does not work when you lose all AC power. You make some very tortured arguments that are heavily dependent on hoping certain things work. You also demand people believe anything you say but sorry, you have done nothing to gain that. The situation at OC has holes in their safety. People want them fixed NOW instead of these constant pointless hair splitting from people who only want to defend the status quo. They need to install a full power back up and back up cooling loop system for the spent fuel pool. Communication and honesty are another animal that needs work but sadly the industry won't do that unless someone forces them by law to act like decent parts of society. The end of the story is the public didn't get enough information and certainly not fast enough out of either entity. If deeper technical disclosure about each plant was made publicly available it would allow people to better address these issues and "it is all safe" statements are generally not helpful.

comment #42712 posted on 2013-01-15 18:32:39 by Hiddencamper in response to comment #42109

Lilly, actually my comments are technically adequate statements and reflect actual nuclear plant operating and safety parameters. I find it somewhat ironic that you say I'm just trying to techno-wabble people, when my comments are consistent with what actually occurred at Oyster Creek and are consistent with press releases from various outlets as well as the NRC's own report. Basically you are saying that people shouldn't listen to me because you dont LIKE what I said, not because I'm wrong. Additionally, for someone who wants correct and accurate information to be publicly available, for you to attack someone's comments for doing that is kind of silly. Also, everything I've said reflects actual conditions at Oyster, along with actual conditions at BWR plants, while your claims require assumptions of failures well beyond what would have occurred. Yes Lilly, it is possible for ALL nuclear power plants to potentially lose all on-site and off-site power, and they would have to use a portable pump to cool the spent fuel pool, how barbaric. Feed and bleed accomplishes the critical safety functions of cooling the fuel and shielding workers. You can't argue that, because it is basic physics. Just because you don't like it doesn't mean it's not correct. This strategy also works without AC power (because it can utilize portable or diesel powered equipment, which Oyster had available). Loss of all on-site and off-site AC power is something that did not occur at Oyster Creek, yet you talk as if it did. Apparently your hypothetical scenarios are more valid than actual scenarios which really occurred. Do you work at Oyster and know something that we dont? The decay heat removal function can easily be accomplished through reactor building HVAC, or as you said, SBGT. Additionally, since the plant was in mode 5 without OPDRVs or fuel moves in effect, secondary containment was not required to be operable. Oyster could have simply opened the reactor building bay doors and vented directly outside. It's allowed under technical specifications and would remove the steam buildup in the reactor building atmosphere accomplishing the decay heat removal function without electrical power! With regards to 22 feet flooding, you need to recognize that is what their safety analysis assumes. OYSTER AND THE NRC ASSUMES LOSS OF SERVICE WATER PUMPS DURING STORM SURGE. But, critical safety functions are still required to function under this condition through other means. The UFSAR could not say they are capable of surviving 22 feet storm surge if they are not capable of it. And finally, there was a LOT of info out there from credible sources regarding the hurricane and Oyster. The NRC had several press releases, and Exelon made several public comments and press releases. They specifically stated they were in the middle of refuelling, they stated they had a portable pump staged to take over the service water function if it was lost, they specifically stated they had reserve fuel and power for their diesels and that there was no threat to the spent fuel pool cooling system as a result. Information was out there, and this most recent incident with Sandy proved that the NRC and nuclear industry are getting better than ever with releases of information to the public.

comment #42668 posted on 2013-01-15 14:57:10 by LillyMunster in response to comment #42625

The NRC was never deceptive about the spent fuel pool cooling vs. ability to put in make up water to prevent fuel uncovering and keep the pool under control. A number of people I know work in the nuclear industry have been busy perpetuating that confusion as did ANS on their website. Instead of speaking clearly they attempted to purposely deceive the public that make up water sources was an actual cooling system. Sadly these same nuclear industry pundits show up on a US government regulatory agency public communications tool and do the same thing here, continuing to try to cloud issues and confuse the public rather than being honest and forthright. No wonder the public has such a level of distrust. These kinds of posts from the industry are problematic and not productive to public discussion, they end up restricting and obstructing the NRC's communication with the public by offering up many times inaccurate or purposely deceptive technical information in an attempt to shut down conversation.

comment #41680 posted on 2013-01-11 15:34:20 by Nancy

Regarding the rising water level.....It is interesting that it was determined that when the water level was greater than 6 feet it would trigger an action but the deck was only six feet and the measurement higher than that had to be taken with a staff gauge. Then when it was determined to have reached 7 feet the the staff gauge which was being used did not measure above 7 feet and they had to resort to another measurement above the water intake pump to get the 7.4 level. All the measuring devices used seem to have been inadequate. If only 7 feet can be measured (rather haphazardly during this event) and the danger zone is 22 feet. shouldn't the plant have measuring equipment that can better handle higher levels?

comment #41678 posted on 2013-01-11 15:32:46 by Joffan

http://pbadupws.nrc.gov/docs/ML1301/ML13010A470.pdf is the direct link., which would have been helpful. The way the Oyster Creek staff handled the change to the water level required to secure the service water pumps was interesting. Clearly this arose from a good understanding of both the requirements of the emergency operating procedure and the capabilities of the plants. All done quickly, correctly and professionally.

comment #41675 posted on 2013-01-11 15:27:40 by Moderator in response to comment #41667

The link to the inspection report has been corrected.

comment #42625 posted on 2013-01-15 12:51:03 by Moderator in response to comment #41669

The NRC is in agreement on the need for a good flow of communications during events like “Superstorm” Sandy. That is why we issued five press releases providing updates during the storm, as well as numerous Tweets and blog posts that also shared the latest details. We will continue to evaluate our approach and seek ways to enhance it in ways that will keep the public better informed. NRC Public Affairs Officers did discuss with reporters the amount of time that existed before a complete loss of cooling to the spent fuel pool would lead to the water there to begin to boil. It is correct that it was estimated at about 28 hours. However, it’s important to note that there were alternate means to provide cooling to the pool even if the service water system, which draws water from the intake canal, was unavailable. This includes using water from an on-site storage tank that holds hundreds of thousands of gallons. Also, the expectation would be that the plant operators would intervene long before it ever reached that point. By that we mean that they would be expected to avail themselves of alternate cooling options through the use of procedures developed to address such scenarios. Neil Sheehan

comment #41668 posted on 2013-01-11 14:25:33 by Ellen Hassett

Thank you for this post, for the special inspection, for continued follow-up reporting, and for collaboration with those concerned.

comment #41669 posted on 2013-01-11 14:39:09 by LillyMunster

This leaves a couple of questions. Why was the public never told of the 10 foot pump height change? Lack of that information caused conflicting public information that 7 foot would cause pump shutdown yet water was at 7.4 but the pumps hadn't been shut down. The public and a number of people in the media has been trying to find out what the actual refueling situation was during Sandy. No information was given leading to speculation that caused more problems than if the NRC or Exelon had told the public the actual situation that 10 assemblies had been moved and the reactor was open. I saw one media report claim the entire core had been offloaded into the pool. Without information to counter things like this the lack of information can cause bigger problems. The public was also not told that cooling for the SFP and the reactor was out for an hour. While technically not a huge deal the public does have a right to know what is going on. Also, if the public had been told the 28 hour to boiling estimate it would have given the public some context. This was a big mess full of miscommunication and an information vacuum. That information vacuum probably worried more people than any of the actual information could. The public also has a right to know in a timely manner.

comment #41667 posted on 2013-01-11 14:24:55 by renodeano

You state the report can be found "here" but it leads you to a site that either does not have it yet or you did not give enough information to the reader to search for it. An IR # would really help in your blog.

comment #42467 posted on 2013-01-15 00:06:36 by James Greenidge in response to comment #42109

Would really like to see some certified engineers back up your definition of a "close call." James Greenidge Queens NY comment #42456 posted on 2013-01-14 22:13:11 by Hiddencamper in response to comment #42109

> And a second reply, there were no safety problems, there was no close call, the plant was still within its reference design basis, and with everything in cold shutdown there was a lot of time to handle any issues. The only thing I could see as lessons learned based on everything I've read, is improvements for organizational preparedness when severe weather is expected. In other words, just as the NRC said, everything was effective and timely.

comment #42455 posted on 2013-01-14 22:11:39 by Hiddencamper in response to comment #42109

> Just a few things to consider. First, 28 hours includes the reactor core, as the fuel gates were removed. This is a VERY long time for the reactor core to reach boiling. Typically 10 days after shutdown the core will boil in less than 1/2 that amount of time. Second, this is only time to boil. With core decay heat it would still take over a day or two prior to water lowering to a level where radiological conditions could be an issue. Third, is there would have been no loss of any other plant systems. Injection systems would have still worked, and engine driven fire pumps would have worked, to provide feed and bleed cooling of the spent fuel pool. Other strategies could have likely been employed. One also needs to consider that Oyster had their FLEX pump (a post-fukushima portable pump), out and was preparing to use it to restore essential service water systems if the service water pumps failed or had to be secured. Fourth, is that it is very easy, in a 28-50+ hour timeframe to get portable pumps or fire trucks onto site to inject to cool the reactor and spent fuel pool. Fifth, is that the plant is designed for a 22 foot flood, which means that assumes a total loss of service water for a period of time, as the pumps would be submerged. In fact Oyster would have shut down their service water pumps if waters threatened them, then after water levels rescinded, they would have cleaned them up and restored them to service, if thats what happened. Boiling is not the end all be all, and there are plenty of options to manage an event like that, especially considering that Oyster had all diesel generators available, and had their combustion turbine (a small power turbine dedicated to the site a couple miles away) available throughout the event.

comment #42953 posted on 2013-01-16 13:08:23 by nuclear guy in response to comment #42625

> Lilly, Oyster Creek had their portable FLEX pump (post-Fukushima enhancement) set up to tie it into the service water system in the event that the essential service water pumps would have had to be secured. It would have restored cooling to reactor closed cooling heat exchangers which then in turn cools spent fuel pool heat exchangers without requiring injection into the spent fuel pool.

comment #41825 posted on 2013-01-12 06:52:13 by James Greenidge

> Neil, your important department has a thankless job! I was dismayed to hear reconfirmed on NYC 1 cable here just how many on-site fatalities and injuries yearly occur at oil and coal and gas electric generating plants around the nation that don't make any news as a matter of "routine" industrial accidents, yet the media are staked out like vultures waiting for just one such incident to occur at a nuclear plant so to blare it out to exaggerate perils and condemn the whole concept. There just aren't kudos enough for the safety record of nuclear plants and the oversight of the NRC!! James Greenidge Queens NY

Improving Communication at the NRC

posted on Wed, 16 Jan 2013 15:13:50 +0000

Lance Rakovan Senior Communication Specialist At the NRC, we do our best to be open and keep the public informed about what actions we are taking and why we are taking them. We are also always open to suggestions on how to improve our communications with the public. On Jan. 23, the NRC will hold a "virtual" public meeting (via webinar and conference call) to discuss potential ways the agency might improve communications. Discussion topics include:

- **Reflections on the NRC's communications since the Fukushima event, including actions the NRC has taken in response**. Since Fukushima, are you getting the information you need involving the NRC and the nuclear industry's progress in implementing lessons learned from the event?
- **Potential actions the NRC might take in the long term to improve stakeholder involvement**. In addition to or instead of its current communication mechanisms, how should the NRC communicate about significant regulatory issues?
- **Ways the NRC could partner with other organizations to improve public communication and education on topics associated with radiological safety.** Which groups might be open to cooperating with the agency on public communications?

- **Non-traditional places/ways the NRC could communicate its message**. Are there unconventional communications channels the NRC is not using that could help get out the agency's message?

Our hope is to get some "out of the box" ideas on ways we can improve howwe communicate with the public. Details about the meeting can be found **here**. Whether or not you can participate in this meeting, please feel free to provide input on any of the topics listed above by commenting to this blog posting. We will incorporate any comments received here into the meeting summary.

Comments

comment #43527 posted on 2013-01-17 17:47:42 by LillyMunster

> 1. Don't take advice from the nuclear industry. :-) BS isn't useful out of a govt. agency. 2. Information during a heightened problem needs to be coming out to the public in a timely manner. NOT just during office hours. That may mean having to create a process to release information during off hours when there is a big situation like a hurricane or a large failure at a plant. Problems don't keep office hours. 3. There needs to be a two layer public communication. Give a clear understandable explanation of the situation and problem so members of the public who do not have a technical understanding can easily understand what is going on. Then include a specific more technical explanation that includes plant systems, what the issue is in standardized terminology and include vital statistics. IE: how many generators are available during an outage situation, how many hours to boil off if cooling was lost, current water levels in a reactor etc etc. so those who DO have the technical understanding can determine what it actually means. This would end lots of the self inflicted problems when there are issues. The public is much smarter and capable of understanding things that some give credit for. 4. There needs to be a system in place where the plant parameter data can be put online publicly in real time if there is an alert or higher situation. Plants already have the ability to send that data to a remote location in real time. Make it public, make a penalty for an operator falsifying that data. That would solve quite a bit of the information vacuum that goes on during these types of incidents. It creates an instant accountability and disclosure to the public. This would also solve quite a bit of needing to relay via NRC staff to the public. 5. Use twitter or other fast communication options for more frequent updates during alerts type situations. Again, solves some of the information vacuum and may create a more streamlined way to get updates out with fewer resources. Faster official bits of information could go a long way towards ending the information vacuum. Information vacuums create rumor, worry and misinformation. 6. Get a handle on the industry interference of the NRC's public communications. The industry, industry consultants, lobbyists, employees etc. all have their own public communications channels, the NRC isn't obligated to give them another one. The FDA would never let Pfizer employees disrupt a public meeting or allow them to harass members of the public attempting to hear or communicate with that agency. Why does the NRC let it happen? Having industry wonks constantly interrupting trying to answer FOR the NRC is inappropriate and not constructive. Some of this behavior borders on intimidation of the public by these people. Think of the member of the public who isn't very assertive but wants to ask a question or hear an answer. Those people should be able to communicate with the NRC without being insulted and verbally bashed over the head by some well known nuclear industry people who do this to squash public conversation and impede communication. 7. Look at newer more efficient communications methods while keeping some low tech ones in place. There will always be a need for the ability of the public to submit in writing or use a phone bridge to hear a meeting. But much of the public is more technically advanced. Any issue that allows public input should allow that input to be submitted online either via email or a web form and treat that the same as it would written submissions. 8. Read up on disaster communications and the use of social media by the public. The Harvard Neiman Report has an excellent series of articles and papers on social media during disasters. It may give those who work in the communications areas some food for thought on how to work with and within social media during times of natural disasters or in the case of a large nuclear accident. http://www.nieman.harvard.edu/reports/issue/100072/Summer-2012.aspx

comment #50102 posted on 2013-01-31 14:52:35 by CaptD

> SanO (aka San Onofre) is now a 1.5 Billion Dollar RED FLAG that illustrates how easy NRC regulations can be gamed (without ANY enforcement penalties) which allow Utilities/Operators to make changes that have enormous implications to safety and the Public Health, with little to N☢ actual oversight, until it is to late! The two basic problems at Fukushima, Japan were that: (1) TEPCO's regulator pushed too much paper instead of being "hands on". (2) TEPCO had total control over what data the public had access to, which prevented any real oversight by the public. The USA cannot afford a Trillion Dollar Eco-Disaster like Fukushima, that is why the NRC needs to "overhaul" how it enforces its current regulations and develop new regulations ASAP to patch all the regulatory holes that now exist! A major first step should be to really open up the entire NRC process to the public, so that true public oversight can take place, instead of the flawed system we now have, as SanO illustrates all too well! As it is now, the public does not have enough access to NRC reports and/or data which prevents all knowledgable people from providing input into the decision making process.

comment #44393 posted on 2013-01-18 08:41:22 by Plumber Kent

> I agree. Please open the website to input , not just comment.

comment #46754 posted on 2013-01-25 13:38:17 by Moderator

> I appreciate all of you who participated in the meeting on January 23rd. In my opinion, a number of inventive and innovative ideas were discussed, along with some very valid reminders of underlying communication principles. The MP3 of the meeting will be made publicly available soon, along with a summary of comments heard at the meeting. Lance Rakovan

comment #50146 posted on 2013-01-31 16:05:12 by Garry Morgan in response to comment #50102

Well stated Cpt D.

comment #50473 posted on 2013-02-01 02:10:17 by Larry Lim

I also think that there needs to be a two way public communication. And that you should give a clear, understandable explanation of the situation and problem so that the public who do not have a technical understanding can easily understand what is going on. And that you should also use more communication methods to the public because I believe there are many available technological resources that you can use nowadays. I think you should hire professional lecturers to show how it’s done would be the quickest way to me. That's all, thank you.

comment #44524 posted on 2013-01-18 15:20:15 by CaptD in response to comment #43527

Stellar comments - Salute

comment #43382 posted on 2013-01-17 14:45:42 by James Greenidge

NCR facilitators: Hire professional lecturers to show how it's done would be the quickest remedy to me. Also, does your training include how to handle media-bait situations such as unreasonable protest demands no business or industry would tolerate and when your findings and conclusions and facts are not what opponents want to hear? Not to sound factious, but if it just boiled down to a pure engineering/management Q&A inquiry session minus passion filler, a meeting wouldn't last more than twenty minutes. I wish chemical plants and gas and oil facilities which have historic fatal incident track records unlike you all offered themselves to be periodically raked over the coals like you guys! James Greenidge Survivor of an Indian Point shout-down circus

comment #43334 posted on 2013-01-17 12:11:30 by Garry Morgan in response to comment #43282

Thank you for attending to the missing post.

comment #43282 posted on 2013-01-17 09:25:47 by Garry Morgan

Deceit and censorship appears to be growing within the NRC. You refused to publish my truthful comment about your security failures, reference "Force-on-Force or Was That a Gunfight at a Nuclear Power Plant" on this blog; then there are the statements from NRC administrators claiming there were no adverse health effects to people as a result of the Fukushima Nuclear Disaster. Apparently the NRC is nothing more than a pawn and propaganda agency for the multinational nuclear industry.

comment #42954 posted on 2013-01-16 13:12:58 by art rosenzweig

open the website to input , not just comment.

comment #50403 posted on 2013-02-01 00:38:04 by Mel Silberberg

From what i can see, the San Onofre restart process has not undergone the traditional peer review process that NRC has used since its inception. This includes peer review by world experts, oversight by the ACRS, and a final process to explain to the affected public that any restart decision was based upon scientific review. The failure of SG tubes at SONGS Unit 2 was not an ordinary occurrence. It is believed to involve such phenomena as fluid elastic instability. If your job, Mr. Rakovan, is public confidence building then a visible, open, and disciplined process is necessary to gain scientific credibility. The comments and effort of respected technical people should not be regarded as coming from anti-nukes. For example, I have worked with one of these technical experts, Dr. Joram Hopenfeld, on a number of technical projects, periodically, over the past 50 years. When Dr. Hopenfeld has concerns, I listen. During the 1980's I managed the NRC RES program on severe accident source terms and through a process as described above, NRC was able to gain broad public and scientific acceptance of accident source term methodology, including the nuclear industry, as well as those in the public sector concerned about nuclear power safety following the TMI-2 accident. The Special NRC Panel on San Onofre Unit 2 Restart falls far short of satisfying the requirements stated above. The public deserves it. The NRC Commissioners should assure the public receives nothing less. Thank you. Mel Silberberg., USNRC-RES retired .

comment #42975 posted on 2013-01-16 14:16:01 by

Open your paradigm. The restrictions NRC puts on public "input" is like shoving everyone into a lavatory with no windows, no wifi, no cell access. It seems to be under what is interpreted as restrictions on the "process", but then why in the world not redefine the "process" and whatever the related "restrictions"? Open it up to a bigger public "room" without the defensive measures. The public is remarkably smart and even knows most of the commission jargon and reference codes. The commission defines input so narrowly that it becomes near meaningless. It's the facile opportunity without the consideration.

comment #43532 posted on 2013-01-17 18:55:22 by Nancy

How paid shills influence debate on the internet http://consciouslifenews.com/paid-internet-shill-shadowy-groups-manipulate-internet-opinion-debate/1147073/

NRC Forms Special San Onofre Review Panel

posted on Thu, 17 Jan 2013 15:58:57 +0000

Victor Dricks Senior Public Affairs Officer Region IV [caption id="attachment_3654" align="alignright" width="300"]

NRC Chairman Allison Macfarlane (second from right) listens as Southern California Edison executive Richard St. Onge (third from right) discusses issues with one of the damaged steam generators at SONGS. The steam generator is in the right foreground.[/caption] The NRC has established a special panel to coordinate the agency's evaluation of Southern California Edison Co.'s proposed plan for restarting its Unit 2 reactor and ensuring that the root causes of problems with the plant's steam generators are identified and addressed. Art Howell, the NRC's Region IV deputy regional administrator, will serve as co-chairman of the panel along with Dan Dorman, deputy director for engineering and corporate support in the Office of Nuclear Reactor Regulation (NRR). Jim Andersen, chief of NRR's Electrical Engineering Branch, will serve as deputy team manager of the San Onofre Nuclear Generating Station (SONGS) Oversight Panel. The panel will ensure that NRC communicates a unified and consistent position in a clear and predictable manner to the licensee, public and other stakeholders, and establishes a record of major regulatory and licensee actions taken and technical issues reviewed, including adequacy of Southern California Edison's corrective actions. The panel also will be responsible for conducting periodic public meetings with the utility and providing a recommendation to senior NRC management regarding restart of SONGS Unit 2. In comments to reporters Monday following a tour of the plant, Chairman Allison Macfarlane said Unit 2 will not be permitted to restart unless the NRC has reasonable assurance it can be operated safely. Other panel members include:

- Ed Roach, chief, Mechanical Vendor Inspection Branch, NRO
- Ryan Lantz, chief, SONGS Project Branch, Region IV
- Greg Werner, inspection & assessment lead, SONGS Project Branch, Region IV
- Nick Taylor, senior project engineer, SONGS Project Branch, Region IV
- Greg Warnick, senior resident inspector, San Onofre Nuclear Generating Station
- Doug Broaddus, chief, SONGS Special Project Branch, NRR
- Randy Hall, project manager, SONGS Special Project Branch, NRR
- Ken Karwoski, senior level advisor, Division of Engineering, NRR
- Michele Evans, director, Division of Operating Reactor Licensing (alternate is Pat Hiland, director, Division of Engineering)

Comments

comment #52762 posted on 2013-02-06 00:22:00 by HelpAllHurtNeverBaba

> Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog SPECIAL EIX CEO/Chairman Awareness Series by HAHN Baba Promoting "Critical Questioning & Investigative Attitude" EIX CEO/Chairman Ted Craver should follow the example of wise decision taken by Brilliant Duke Energy CEO Jim Rogers and shutdown the Terminally Sick San Onofre just like Terminally Sick Crystal River. The EIX/SCE should review alternatives to replace the power produced by San Onofre by construction of new, state-of-the-art, natural gas-fueled/solar 50-100 MW plants installed throughout the grid. The decision to retire the Terminally Sick San Onofre nuclear plant would be the best in overall interests of Southern California Customers/Public, EIX Investors/Shareholders, the State of California, CPUC and NRC Region IV. The decision would be very difficult, but it would be the right economical and safety choice and politically popular.

comment #52718 posted on 2013-02-05 23:19:21 by HelpAllHurtNeverBaba

> Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog SPECIAL Public/NRC/SCE Awareness Series by HAHN Baba Courtesy of The DAB Safety Team A total of 24 Alloy 690, chrome-plated retainer bars welded to the retaining bars are provided to prevent AVB structure displacement during SG fabrication and during a limiting design basis accident such as a main steam line break. The retainer bars anchor the AVB structure to the tubes, but are designed not to contact the tubes under operating conditions. As shown in Section 5.5, Edison response to NRR RAI #15, SCE states, "The limited vibration amplitude of the tubes and retainer bar, combined with stabilizer development, prevents developing wear displacement /wear geometry that could severe any of the tubes adjacent to the retainer bars, either in the short term or long term." This statement is unacceptable, because the conclusions appear to be drawn without any publicly available SCE auditable scientific/testing data and structural, materials engineering and thermal-hydraulic calculations. The structural integrity of SONGS Unit 2 replacement steam generators degraded retainer bar system welds, retainer bars, stabilized and non-stabilized plugged tubes to withstand combined loads that result from postulated accident

conditions events as assumed in the RSG Design/FSAR Analysis has not been demonstrated. This includes a design basis earthquake (DBE) in combination with a LOCA (multiple SG tube leak and/or rupture events due to FEI caused by U-tube bundle uncovery) and MSLB (high energy flashing feedwater jet impingement and loose parts causing multiple tube leak and/or rupture events due to SG depressurization). Like John Large says,"Put another way, the extensive and rapid rates of tube wear experience at the SONGS Unit 2 and Unit 3 RSGs, have necessitated an extensive raft of analysis, assessments and projections to qualify, or otherwise, that Unit 2 is fit for purpose. Not only is this prequalifying work unique to the San Onofre nuclear plant, much of it has never been undertaken before so, it follows, its inclusion in safety considerations must be a new and hitherto unconsidered component now required to be incorporated into an updated version of the FSAR." San Onofre NRC AIT Report, SCE Unit 3 Cause Evaluation, SCE Unit 2 Return to Service Reports, SCE Response to NRR RAIs, San Onofre Special Tube Inspection Reports and 10 CFR 50.59/FSAR Justifications need to be thoroughly reviewed and a GAP Analysis prepared by NRC NRR, Civil, Mechanical, Chemical, Materials, Structural, Electrical/I&C, T/H Engineers, Computer Modeling and San Onofre Special NRC Panel Members. San Onofre Special NRC Panel Members need to make accurate and precise engineering decisions based on validated and auditable facts in accordance with Honorable and Respected Dr. McFarlane's High Standards. These decisions have to be made without any political/financial/time pressures from EIX/SCE Officers, CPUC Chairman, NRC Commissioners, Pro-SCE Politicians, Attorneys or Industry Lobbyists. Ex NRC Branch Chiefs (Dr. Joram Hopenfeld, Mel Silberberg, etc.), Anonymous "Critical Questioning & Investigative Attitude" Genius NRC Branch Chief, US Public, San Onofre Workers and Southern Californians would appreciate San Onofre Special NRC Panel Members "Critical Questioning & Investigative Attitude", True, Unbiased and Diligent Public Safety Efforts.

comment #46909 posted on 2013-01-26 00:49:19 by HelpAllHurtNeverBaba in response to comment #45344

Hi Mr. Silberberg, Brilliant Question And Great Recommendation... My Salute ... HAHN BABA SONGS RSG Failure Root Cause - Lack of "Critical questioning and an investigative attitude" by SCE, MHI and NRC Region IV A NRC Branch Chief gifted with MIT Intelligence, Intuition and a Sixth Sense, who is an acquantaince of mine, told me at an Industry Conference, "Sir to resolve any complex technical problem and understand unclear regulations, you have to, 'Read and reread in between the lines', use, 'Critical questioning and an investigative attitude' and 'Solid teamwork & alignment." Thanks to NRC for posting this comment.. HAHN BABA

comment #46433 posted on 2013-01-24 19:55:43 by CaptD

San Onofre is rated by the Institute of Nuclear Operations (INPO) as an INPO 4 Plant (The Worst Nuclear Plant Rating) and it should also should be rated in NRC Region IV Response Column V (Worst rating) and not in the NRC Response Column I (Best Nuclear Plant Rating). San Onofre is the worst nuclear plant in the country with the worst safety record, worst retaliation record, an INPO 4 rating and it is a mockery to place it in NRC Response Column I. NRC Region IV by listing San Onofre in NRC Response Column I, is putting its credibility on line and is displaying clear trends of collusion with SCE. It would be informative to learn who made the decision on San Onofre's current ranking and why... If the NRC San Onofre Special Review Panel wants to be welcomed by Southern Californians at their upcoming February 12 Public Meeting with SCE , the NRC needs to change San Onofre's rating to NRC Response Column V, which will reflect current reality instead of just wishful thinking. Definitions of NRC Response Columns: Column I - All performance indicators and NRC inspection findings are GREEN Column II - No more than two WHITE inputs in different cornerstones. Cornerstone objectives fully met. Column III - One degraded cornerstone (two WHITE inputs or one YELLOW input or three WHITE inputs in any strategic area). Cornerstone objectives met with minimal reduction in safety margin. Column IV - Repetitive degraded cornerstone, multiple degraded cornerstones, or multiple YELLOW inputs, or one RED input. Cornerstone objectives met with long-standing issues or significant reduction in safety margin. Response at NRC Agency level • Executive Director for Operations to hold public meeting with senior utility management • Utility develops performance improvement plan with NRC oversight • NRC team inspection focused on cause of degraded performance • Demand for Information, Confirmatory Action Letter Column V. Unacceptable Performance, Unacceptable reduction in safety margin Response at NRC Agency level •Plant not permitted to operate

comment #45124 posted on 2013-01-20 21:26:49 by HelpAllHurtNeverBaba

Just trying to help the NRC San Onofre Special Panel with some of the facts: 1. San Onofre Emergency Preparedness DEP Indicator Value is consistently amongst the lowest in the US Nuclear Power Plants, 2. The Shift Manager Training Guru was on duty at the time of San Onofre Unit 3 Accident, so the reactor was shutdown in a timely and safe manner. Southern Californians were lucky, 3. The other best known Shift Manager resigned due to differences with plant management, 4. The best known Station and Corporate Emergency Directors have retired, 5. The other Shift Managers, Station and Corporate Emergency Directors record of accomplishment is for NRC San Onofre Special Panel to judge, 6. The Manager of Plant Operations is very knowledgeable, and 7. Therefore, the probability of success to avert another potential accident due to Restart of Defectively-Designed and Degraded Unit 2 Replacement Steam Generators at 70% power is 50% based upon who is on duty at the time of the accident (due to a design bases main steam line break or other anticipated operational occurrences). Therefore, the decision of NRC San Onofre Special Panel should take into account the above facts. Thanks for posting.

comment #45122 posted on 2013-01-20 20:47:53 by HelpAllHurtNeverBaba

EPRI, NRC, Westinghouse, AREVA and MHI ATHOS thermal-hydraulic computer models cannot accurately account for all the mechanical and structural unknowns, and extremely narrow tube-to-tube clearance differences, which would very likely govern the catastrophic tube-to-tube wear (fluid elastic instability) in San Onofre Unit 2 during a main steam line break or other anticipated operational occurrences at 70% power. Computer Modeling predictions are as good as the input based on the as-built hot pressurized U-Tube Bundle Anti-Vibration Structure behavior, which nobody knows. John Large, Internationally Known Scientist and Chartered Nuclear Engineer from London says about the SONGS Unit 2 Replacement Steam Generators (RSGs) AVB Structure, "It impossible to reliably predict the effectiveness of the many thousands of AVB contact points for when the tube bundle is in a hot, pressurized operational state. The combination of the omission of the in-plane AVB restraints, the unique in-plane activity levels of the SONGS RSGs, together the very demanding interpretation of the remote probe data from the cold and depressurized tube inspection, render forecasting the wear of the tubes and many thousands of restraint components when in hot and pressurized service very challenging indeed."

comment #46003 posted on 2013-01-23 18:22:11 by HelpAllHurtNeverBaba in response to comment #45861

Mr. Dricks. Would you please make the documents containing the findings of these experts public by posting them on the NRC website, because these are NRC documents and not Licensee documents. Please do it to assure the public of NRC independent conclusions, because public pays all the bills for the government via taxes. Thanks.

comment #46707 posted on 2013-01-25 11:26:45 by CaptD in response to comment #46249

SanO is now a 1.5 Billion Dollar RED FLAG that illustrates how easy NRC regulations can be gamed (without ANY enforcement penalties) which allow Utilities/Operators to make changes that have enormous implications to safety and the Public Health, with little to N☢ actual oversight, until it is to late! The two basic problems at Fukushima, Japan were that: (1) TEPCO's regulator pushed too much paper instead of being "hands on". (2) TEPCO had total control over what data the public had access to, which prevented any real oversight by the public. The USA cannot afford a Trillion Dollar Eco-Disaster like Fukushima, that is why the NRC needs to "overhaul" how it enforces its current regulations and develop new regulations ASAP to patch all the regulatory holes that now exist! A major first step should be to really open up the entire NRC process to the public, so that true public oversight can take place, instead of the flawed system we now have, as SanO illustrates all to well! As it is now, the public does not have enough access to NRC reports and/or data which prevents all knowledgable people from providing input into the decision making process.

comment #46697 posted on 2013-01-25 10:53:29 by CaptD in response to comment #46572

Great comment - I'm looking forward to additional posts from you - Salute! Getting far more qualified people involved and especially professionals from outside of the NRC and most importantly from outside of Region IV, is the first step toward answering basic reactor fatigue safety questions that we now know, affect the entire US Nuclear Fleet. If we learned nothing else from the Fukushima tragedy, we now know that when it come to reactor safety, the widest possible public review can only help insure against future nuclear accidents. Since you are a QA professional I urge you to read : "Press Release 13-01-22 ATHOS Validity Questioned, Qualifying Investigation Required" Validity of ATHOS computer model requires NRR Qualifying Investigation. (3 Pages) https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=1ltCb57ciXRaOkhK1rhc2BaB0ACXf7MwcSDZZyEAkFDI or this one for much more in-depth technical information: "Response to NRR RAI #32 - Technical" The SCE cannot provide an ACCEPTABLE operational assessment to the NRR, therefore NO RESTART IS POSSIBLE and here ARE THE TECHNICAL REASONS WHY (50 Pages) https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=0BweZ3c0aFXcFX05DMWxKNmZXUTA and/or the even the longer paper: "SCE NRC Presentation analysis + 14 Questions 12-12-17" Technical document includes 14 questions affecting US Reactor SAFETY, that the NRC, NRR and RES Regulators need to ask SCE at their 12/18/12 NRR/RES Meeting. (78 Pages) https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=0BweZ3c0aFXcFRzBqZUJROWRYNlE

comment #46305 posted on 2013-01-24 13:44:21 by Moderator in response to comment #45861

ACRS briefings on event-driven issues typically occur after the NRC staff has finished with inspection and oversight activities, which continue with SONGS. The ACRS main number is 301-415-7360 Edwin Hackett Executive Director ACRS

comment #45000 posted on 2013-01-20 13:12:16 by Hiddencamper in response to comment #44456

The 50.59 is required for ANY change in the plant. The 50.59 was performed. 50.59 is what says you don't need a license amendment. I think you're talking about the 50.90 process for a license amendment.

comment #45137 posted on 2013-01-21 00:37:55 by HelpAllHurtNeverBaba

Just trying to help the NRC San Onofre Special Panel with some of the facts: 1. NRC Augmented Inspection Team Report and SCE Cause evaluations on both San Onofre Unit 3 and 2 FEI are still unresolved and open based: A. ATHOS limitations disputed by John Large, Arnie Gundersen, Academic Research Scholars and DAB Safety Team, B. Insufficient tube-to-AVB contact forces on Unit 3 disputed by DAB Safety Team, Westinghouse, MHI, AREVA and John Large, and C. Operational Factors based on the information from San Onofre Plant Data disputed by SONGS Root Cause Team Member and DAB Safety Team. Therefore, the decision of NRC San Onofre Special Panel should take into account the above facts. Thanks to the NRC Moderator for posting this information.

comment #48685 posted on 2013-01-29 15:07:09 by HelpAllHurtNeverBaba in response to comment #46249

Special Public Awareness Series – SONGS $1Billion Dollar Radiation Steaming Crucibles Unbiased and Factual Information provided for the benefit of NRC San Onofre Special Panel Addressed To: Ryan Lantz, Brian Benney, Randy Hall, Edwin Hackett, Dan Dorman, Victor Dricks Good Moring Mr. Dricks, SONGS Insider Information from Anonymous Sources and From DAB Safety team for SONGS Special Onofre Team - Response appreciated from the San Onofre Special Panel A NRC Branch Chief gifted with MIT Intelligence, Intuition and a Sixth Sense, who is an acquaintance of mine, told me at an Industry Conference, "Sir, to resolve any complex technical problem and understand unclear regulations, you have to, 'Read and reread in between the lines', use, 'Critical questioning and an investigative attitude' and 'Solid teamwork & alignment." Allegation – NRC AIT Report Incomplete, Inconclusive, Inconsistent and Unacceptable SONGS UNIT 3 RSG REAL ROOT CAUSE: Lack of "Critical Questioning & Investigative Attitude" by SCE, MHI, NRC Region IV and the AIT Members. NOTE: ATHOS Modeling results are not reliable, because the results by NRC AIT Team, Westinghouse, MHI, AREVA and Independent Experts show that fluid elastic instability occurred both in Units 3 and 2. The investigations in the Root cause of SONGS Unit 3 FEI regarding computer modeling have not been completed by NRC AIT Team, SCE and MHI. As shown in item 3 below, FEI did not occur in Unit 2 according to DAB Safety Team and Westinghouse. As also shown in other DAB Safety Team reports, FEI was not caused in Unit 3 by tube-to AVB gaps as bogusly claimed by NRC AIT Team and SCE. This is consistent with the findings of Westinghouse, AREVA, MHI, John Large and SONGS Anonymous Insiders. The concerns raised by Dr. Hopenfeld are extremely important safety issues. As the ACRS stated: • Steam generators constitute more than 50% of the surface area of the primary pressure boundary in a pressurized water reactor. • Unlike other parts of the reactor pressure boundary, the barrier to fission product release provided by the steam generator tubes is not reinforced by the reactor containment as an additional barrier." • Leakage of primary coolant through openings in the steam generator tubes could deplete the inventory of water available for the long-term cooling of the core in the event of an accident. In the decade since Dr. Hopenfeld first raised his safety concerns, the NRC has allowed many nuclear plants to continue operating nuclear power plants with literally thousands of steam generator tubes that are known to be fatigue cracked! The ACRS concluded that the NRC staff made these regulatory decisions using incomplete and inaccurate information. After receiving the ACRS's report, the NRC staff considered Hopenfeld's concerns "resolved" even though it had taken no action to address the numerous recommendations in the ACRS report. Mel Silberberg January 21, 2013 at 6:31 pm US NRC Blog I am disappointed in the composition of the special panel! Where is the representation from NRC-RES? The issues at SONGS involve thermal hydraulics and material science. The NRC-RES and its contractors are experts in these areas. The Office of Research was created by the Congress for such situations. Two RES staff covering these disciplines and one or two consultants, serving as peer-reviewers. Perhaps there needs to be a separate peer review. Public confidence can only be gained using logical, informed measures as I described above. Mel Silberberg, NRC-RES, Retired [Chief, Severe Accident Research Branch; Waste Management Branch. 1. Changes in SONGS RSGs from Original CE OSGs In the SONGS RSGs: the number of tubes were increased by 377 and made > 7 inches taller to achieve 11% increase in Heat transfer Area of Tubes to increase 24MWt per RSG, tube wall thickness was reduced from 0.048 inches to 0.043 to pump more reactor coolant through the tubes, moisture content was reduced from 0.2% to 0.1% to improve SG performance , secondary pressure was reduced from 900 psi to 833 psi to push more heat from the reactor coolant to the feedwater, RCS Volume was increased from 1895 cubic feet to 2003 cubic feet, RCS Flow was increased from 198,000 gpm to 209,000 gpm, feedwater flow was increased from 7.4 million pound per hour to 7.6 million pound per hour and AVBs were not designed to prevent against adverse effects of fluid elastic instability (In-plane vibrations, Tube-to-Tube wear, steam dry-outs). These unapproved and unanalyzed changes were claimed to be a conservative move and improvements in the RSGs from OSGs under a "like for Like" change. No mixing baffles were added in the SONGS RSGs to improve the T/H Performance and eliminate dead zones in the RSGs. Palo Verde made similar changes to their RSGs under a 50.90 License Amendment. PVNGS Generators are running after 10 years with very little tube plugging whereas the above changes in SONGS RSGs destroyed Unit 3 and crippled Unit 2 RSGs. These fatal changes definitely: a) Caused a significant increase in the probability or consequences of an accident previously evaluated (SGTR) and, b) involve a significant reduction in a margin of safety – Failure of 8 Unit 3 SG Tubes under MSLB test conditions and significant TTW > 35% of ~381 tubes in Unit 3 RSGs. 2. Problems in SONGS Original CE Steam Generators: In the Original 2001 Power Uprate Application (NRC ADAMS Accession Number ML010950020), "Proposed Change Number NPF-10115-514 Increase in Reactor Power to 3438 MWt San Onofre Nuclear Generating Station Units 2 and 3", SCE stated " By the above reference Southern California Edison (SCE) submitted Amendment Application Numbers 207 and 192 to the facility operating licenses for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively, to increase the licensed reactor thermal power level to 3438 MWt. At 100% power operation, steam generator pressures typically vary between 800 psia and 815 psia, compared to the original nominal design operating pressure of 900 psia. Wear at tube support structures is a known degradation mechanism at SONGS. At SONGS, rapid wear was observed on tubes surrounding the stay cylinder in the center of the steam generator during the first cycle of operation. Many tubes in the most susceptible region around the stay cylinder have been preventively plugged. The first preventive plugging was done after 0.7 EFPY of operation. The preventively plugged region was expanded during the Cycle 3 outage. Typical active wear in CE designed steam generators has occurred at the support structures in the upper bundle region of the steam generator. These supports consist of diagonal straps (frequently called bat wings) and vertical strap supports. This currently active wear mechanism is influenced by both flow velocities and tube to support gap wear. The variable influenced by the proposed uprate is the inner bundle flow velocities. The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit, i.e., small perturbations of steam pressure or circulation ratio will diminish rather than grow in amplitude. The damping factors remain highly negative, at a level comparable to the current design, for all cases. Thus, the steam generators remain hydrodynamically stable for all uprate cases.Based on a projected increase of 2.3% in the secondary side fluid velocity, normal operation flow induced vibration analysis is impacted by the velocity increase. Current analysis considered that tubes with more than one consecutive inactive eggcrate were staked and plugged, and two nonconsecutive inactive eggcrates are acceptable. The Stability Ratio (SR) is defined as: SR = Veff/Vcr, where, Veff= effective velocity, Vcr = critical velocity; and Values of SR 99%), known as "NO Effective Thin Tube Film Damping." Thin film damping refers to the tendency of the steam inside the generators to create a thin film of water between the RSG tubes and the support structures and each other. That film is enough to help keep the tubes from vibrating with large amplitudes, hitting other tubes violently, and to protect the Anti-Vibration Bar support structures and maintain the tube-to-AVB gaps and contact forces. These adverse conditions in Unit 2 at 70% power operation (RTP) with the present defective design and degraded RSGs, known as fluid elastic instability (Tube-to-Tube Wear, or TTW) can lead to rapid U-tube failure from fatigue or tube-to-tube wear in Unit 2 due to a main steam line break as seen in Unit 3's RSG's. In summary, FEI is a phenomenon where due to San Onofre RSGs design intended for high steam flows causes the tubes to vibrate with increasingly larger amplitudes due to the fluid effective flow velocity exceeding its specific limit (critical velocity) for a given tube and its supporting conditions and a given thermal hydraulic environment. This occurs when the amount of energy imparted on the tube by the fluid is greater than the amount of energy that the tube can dissipate back to the fluid and to the supports. The lack of Nucleate boiling on the tube surface or absence of water is found to have a destabilizing effect on fluid-elastic stability. C.2 – Unit 2 FEI Conflicting Operational Data • NRC AIT Report SG Secondary U2/3 Pressure Range 833 – 942 psi • SCE RCE SG Secondary U2/3 Pressure – 833 psi • RCE Team

Anonymous Member – Unit 2 SG Secondary Pressure 863 psi • SONGS SG System Description Unit 2 SG Pressure Range 892 – 942 psi • Westinghouse OA SG Secondary U2/3 Pressure ~ 838 psi, Void Fraction 99.55% • SCE Enclosure 2, MHI ATHOS results – U2/3 Void Fraction 99.6% • SCE Enclosure 2, Independent Expert results – ATHOS U2/3 Void Fraction 99.4% • DAB Safety Team SG Secondary U2 Pressure 863 -942 psi, Void Fraction 96-98% • SONGS Plant Daily Briefing Unit 3 Electrical Generation – 1186 MWe • SONGS Plant Daily Briefing Unit 2 Electrical Generation – 1183 MWe C.3 – Unit 2 FEI Conclusions C.3.1 – NRC AIT Report – Operational Differences between U2/3 – The NRC analysis indicated a correlation with the tube-to-tube wear based on a combination of high void fraction and high steam velocities. It should be noted that the traditional forcing function, fluid velocity squared times density, does not show good agreement with the tube-to-tube wear patterns. This indicated that the high quality steam fluid velocities and high void fraction may be sufficiently high to cause conditions in the generators conducive for onset of fluid-elastic instability. The ATHOS code predicted regions of high void fraction and high steam velocities are super-imposed with tube-to-tube wear indications from Unit 3 steam generator 3E0-88 The above analyses apply equally to Units 2 and 3, so it does not explain why the accelerated fluid-elastic instability wear damage was significantly greater in Unit 3steam generators. The result of the independent NRC thermal-hydraulic analysis indicated that differences in the actual operation between units and/or individual steam generators had an insignificant impact on the results and in fact, the team did not identify any changes in steam velocities or void fractions that could attribute to the differences in tube wear between the units or steam generators. C.3.2 – SCE Unit 2 Restart Report Enclosure 2 Conclusions – Because of the similarities in design between the Unit 2 and 3 RSGs, it was concluded that FEI in the in-plane direction was also the cause of the TTW in Unit 2. C.3.3 – SCE U2 FEI SONGS RCE Team Anonymous Member Conclusions – FEI did not occur in Unit 2. C.3.4 – Westinghouse OA Conclusions: (a) An evaluation of the tube-to-tube wear reported in two tubes in SG 2E089 showed that, most likely, the wear did not result from in-plane vibration of the tubes since all available eddy current data clearly support the analytical results that in-plane vibration could not have occurred in these tubes, and (b) Operational data – Westinghouse ATHOS Model shows no operational differences in Units 2 & 3 (void fraction ~99.6%) and then Westinghouse says in (a) above that FEI did not occur in Unit 2. Westinghouse is contradicting its own statement. C.3.5 – AREVA OA Conclusions – Based on the extremely comprehensive evaluation of both Units, supplemented by thermal hydraulic and FIV analysis, assuming, a priori, that TTW via in-plane fluid-elastic instability cannot develop in Unit 2 would be inappropriate. C.3.6 – John Large States, “I note here that there are three clear conflicts of findings between the OAs: From AREVA that AVB-to-tube and TTW result from in-plane FEI, contrasted to Westinghouse that there is no in-plane FEI but most probably it was out-of-plane FEI, and from MHI that certain AVB-to-tube wear results in the absence of in-plane FEI from just turbulent flow. My opinion is that such conflicting disagreement over the cause of TTW reflects poorly on the depth of understanding of the crucially important FEI issue by each of these SCE consultants and the designer/manufacturer of the RSGs.” C.3.7 – DAB Safety Team Conclusions – Due to higher SG pressure (Range 863 – 942 psi) and lower thermal megawatts as compared to Unit 3, FEI did not occur in Unit 2. This is consistent with the position of RCE Team Anonymous Member. The NRC AIT Report, SCE, Westinghouse, MHI, Independent Expert and AREVA conclusions on Unit 2 FEI are Contradicting, Confusing, Inconclusive, Full of Smoking Mirrors, Inconsistent and Unacceptable PROBBABLE ROOT CAUSE: Lack of “Critical Questioning & Investigative Attitude” of SCE Supplied Operational Data by Westinghouse, AREVA, MHI and Other World’s Leading Experts

comment #48683 posted on 2013-01-29 14:55:12 by Mel Silberberg in response to comment #48656

Victor: Inspection Reports are only one facet of the problem, no question. However,understanding the reasons for the fluid instability, possible cavitation corrosion effects, etc.are phenomena which require evaluation by T/H as well as materials experts, with appropriate oversight by the ACRS. The SCE, the nuclear industry, the NRC and the public need assurance, not educated guesses. I have not seen a bona fide attempt to understand resolve the issue such that all can be alert to potential problems. I still remain puzzled as to why the ACRS [at least one of the Subcommittees]. i am trying to reach the ACRS Exec. Director to discuss this point. Thank you. Mel Silberberg

comment #50107 posted on 2013-01-31 14:57:03 by CaptD

Happy Anniversary! I predict that time will show that a nuclear accident (not a nuclear incident) was narrowly avoided at SanO on January 31, 2012 only because of shear luck, due to the timing of the discovery of Edison's poorly in-house designed replacement steam generators (RSG). Had that Unit 3 tube been just a tiny bit stronger and not leaked when it did; then with both Unit 2 & 3 back online, if a main steam line break or something similar occurred, we now know that it would have probably resulted in the complete venting of the core coolant within minutes, and we all know what that means... SanO is now a 1.5 Billion Dollar RED FLAG that illustrates how easy NRC regulations can be gamed (without ANY enforcement penalties) which allow Utilities/Operators to make changes that have enormous implications to safety and the Public Health, with little to N☢ actual oversight, until it is to late! The two basic problems at Fukushima, Japan were that: (1) TEPCO's regulator pushed too much paper instead of being "hands on". (2) TEPCO had total control over what data the public had access to, which prevented any real oversight by the public. The USA cannot afford a Trillion Dollar Eco-Disaster like Fukushima, that is why the NRC needs to "overhaul" how it enforces its current regulations and develop new regulations ASAP to patch all the regulatory holes that now exist! The first step is to really open up the entire NRC process to the public, so that true public oversight can take place, instead of the flawed system we now have, as SanO illustrates all too well! As it is now, the public does not have enough access to NRC documents, reports and/or data which prevents all knowledgable people from providing true input into the decision making process. Or said another way, we cannot afford to have a Trillion Dollar Eco-Disaster in the USA for any reason and that includes GREED... What we don't know can indeed hurt US, especially if it is radioactive!

comment #50097 posted on 2013-01-31 14:48:14 by CaptD

If a picture is worth a thousand words, then a video is worth an AIT Report... The latest video from Friends of the Earth US: "No way out" http://www.youtube.com/watch?v=do7462blgdE&feature=youtu.be

comment #44989 posted on 2013-01-20 12:17:10 by maccad

@HelpAllHurtNeverBaba Thanks for your interesting and very-well written comment!

comment #45647 posted on 2013-01-22 16:53:22 by Moderator in response to comment #45540

The “Corporate Support” portion of my title refers to oversight of budget and staffing for the NRC’s program for licensing and oversight activities involving operating reactors. Most NRR staff members are actively involved in public support and outreach through, among other things, timely posting of public records to the agency’s document management system, ADAMS, and through planning and participating in public meetings on many diverse topics. In addition, NRR is supported in public outreach by other offices, including the role of the Office of Public Affairs in providing social media such as this blog. Dan Dorman

comment #50749 posted on 2013-02-01 13:20:47 by CaptD in response to comment #50257

I think that California does not NEED any nuclear power plants since they have plenty of spare capacity without relying on nuclear generation! Add in the RISK of an Earthquake and/or a Tsunami and the fact that California has plenty of sunshine, not to mention the possibility of off shore (out of sight of those on land) wind generation and you realize that California could become an Energy exporter, all without any nuclear generation or the massive amount of waste they create! The only thing keeping California from going Non-Nuclear is the "Public Utilities" which now have a strangle-hold on the states Political Leadership and their Utility Regulators...

comment #45687 posted on 2013-01-22 22:26:04 by HelpAllHurtNeverBaba in response to comment #44989

Hi, Thanks for your very kind comments. I am just a very average person, lucky to be working with a very dedicated and highly technical team trying to establish facts about San Onofre.

comment #50257 posted on 2013-01-31 20:25:47 by HelpAllHurtNeverBaba

Southern Californians need safe, affordable, reliable, well managed, well maintained and excellently operated nuclear power plants, where workers are free for raising nuclear safety and personnel concerns and Rate Payers, Regulators, Politicians and News Media are Proud. San Onofre does not meet any of the above listed criteria. With that said, SCE has to meet all of the above criteria or Decommission San Onofre. Thanks To the NRC Moderator for posting this blog... HAHN Baba

comment #50280 posted on 2013-01-31 21:40:45 by HelpAllHurtNeverBaba

San Onofre Special Public/NRC/SCE Awareness Series by HAHN Baba Courtesy of DAB Safety Team Press Release - The DAB Safety Team: January 31, 2013 Four More Statements From NRC Region IV Augmented Inspection Team (AIT) That Require A Nuclear Reactor Regulation (NRR) Investigation And Resolution. The DAB Safety Team Has Transmitted The Following Request To The Offices Of Chairman Of The NRC, The California Attorney General and Senator Barbara Boxer’s Committee on Environment and Public Works (EPW). 1. NRC AIT in its report dated November 09, 2012 (Re: NRC ADAMS Library Accession Number ML 2012010 - Unresolved Item 05000362/2012007-03, “Evaluation of Unit 3 Vibration and Loose Parts Monitoring System Alarms (V&LPM)”) closed the referenced item by stating that, “The inspectors determined that the licensee properly responded to and evaluated the alarms and followed the applicable station alarm procedures and vendor recommendations. Subsequently, the licensee requested from the vendor an in-depth evaluation of the available acoustical data, which was documented in Nuclear Notification NN 201818719. This evaluation established the likely source of the alarms. The results were inconclusive because of limitations with the monitoring system. Specifically, because of sensor locations (lower portion of the steam generator below the tube sheet in the support structure) and sensitivity, it was not possible to determine the exact source of the Unit 3 alarms. Westinghouse engineering personnel performed an evaluation (Evaluation 201818719-SPT-2) of acoustical data and determined from the shape and intensity of the particular responses that the acoustic source was not likely from the upper bundle of the replacement steam generator or related to the tube-to-tube wear. The licensee (SCE) is considering additional sensor locations which are not required, but may help with monitoring the upper bundle region of the steam generator during power operation. The results of this additional monitoring and increased sensor sensitivity may provide the licensee with a potential means to monitor for tube-to-tube degradation.” (The wonders of of this improved version of V&LPM system related to of tube-tube wear as claimed by AIT Team and SCE and questioned by NRR as NO detection capability below). According to the December 18, 2012 SCE NRC Public meeting Press and Webcast Reports, Edison officials came under sharp questioning about the Vibration and Loose Parts Monitoring System monitors at a U.S. Nuclear Regulatory Commission panel meeting in Maryland. Richard Stattel of the NRC’s Nuclear Reactor Regulation (NRR) Instrumentation Branch told the Edison Officials in a roaring and loud voice on an international live web cast, “The equipment could not do the job described by the company or provide additional safety if the plant is restarted. The instrumentation that you're proposing ... does not appear to be capable of detecting the conditions that would lead to actual tube wear." Edison depicted the equipment in its restart plan as an important safety measure "but it doesn't appear to do that.” See the DAB Safety Team’s Press Release + 12-12-28 Thirty Alarms Demonstrates SONGS Unsafe for details on this subject. DAB Safety Team Comments: The NRR is saying loud and clear that both NRC AIT and SCE Engineers need to understand the basic functions of “Safety-Grade” Instrumentation and the concept of “tube-to-tube” wear (Fluid Elastic Instability). Since there are no means of monitoring tube wall thinning while the plant is in service, the risk of tube burst is wholly dependent upon the accuracy and reliability of SCE’s “Safety-Grade” Instrumentation. The DAB Safety Team has stated earlier that NRC AIT Report is just a replication of SCE Root Cause Evaluation and not a true assessment by an Independent Regulator tasked with ensuring Public Safety. On December 21, 2012, the US Nuclear Regulatory Commission (NRC) blog posted a letter from Chairman Macfarlane titled, “A Visit to Japan: Reflections from the Chairman.” She said, “Regulators may need to be “buffered” from political winds, but they need to be fully subjected to the pressure of scientific and engineering truth and cannot be allowed to make decisions or order actions that are "independent" of facts." According to the March 16, 2012 Press reports, Senators Barbara Boxer (D-CA), Chairman of the Senate Environment and Public Works Committee (EPW), and Dianne Feinstein (D-CA) sent a letter to the Chairman of the Nuclear Regulatory Commission (NRC), Dr. Gregory Jaczko, calling on the NRC to perform a thorough inspection at the San Onofre plant, located in San Clemente. The collusion and casual relationship between NRC AIT Team and SCE requires an Investigation by the Offices of NRC Chairman and Honorable and respected Senator Barbara Boxer to determine the impact on both future US reactor operations and emergency preparedness planning. This investigation by the AIT does not meet the Honorable and Respected NRC Chairman's Standards. 2. NRC AIT in its report dated November 11, 2012 (Re: Unresolved Item 05000362/2012007-03, "Evaluation of Retainer Bars Vibration during the Original Design of the Replacement Steam Generators") closed the referenced item by stating that, "The inspectors determined that the licensee's failure to verify the adequacy of the retainer bar design as required by SONGS Procedure SO123-XXIV-37.8.26 was of very low safety significance (Green) based on NRC Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," because the finding did not involve a degraded steam generator tube condition where one tube could not sustain 3 times the differential pressure across a tube during normal full power, steady state operation and none of the replacement steam generators violated the "accident leakage" performance criterion in plant Technical Specifications as a result of the retainer bar vibrations. The licensee also implemented actions to inspect all affected tubes in Unit 2 and 3 and remove from service all those tubes surrounding the smaller retainer bars that could wear due to vibration of the retainer bar. Because this violation has been determined to be of very low safety significance (Green) and has been entered in the licensee's corrective action program as SONGS Nuclear Notification (NN) 201843216, it will be dispositioned as a non-cited violation in accordance with Section 2.3.2 of the NRC's Enforcement Policy." John Large, internationally known Consulting Engineer, Chartered Engineer, Fellow of the Institution of Mechanical Engineers, Graduate Member of the Institution Civil Engineers, Learned Member of the Nuclear Institute and a Fellow of the Royal Society of Arts states concerning SONGS Restart Unit 2 in his testimony to the Atomic Safety Licensing Board, "In October 2012 MHI reported directly to the NRC safety concerns about the retainer bars: The Steam Generator tube wear adjacent to the retainer bars was identified as creating a potential safety hazard. The maximum wear depth is 90% of the tube thickness. The cause of the tube wear has been determined to be the retainer bars' random flow-induced vibration caused by the secondary fluid exiting the tube bundle. Since the retainer bar has a low natural frequency, the bar vibrates with a large amplitude. This type tube wear could have an adverse effect on the structural integrity of the tubes, which are part of the pressure boundary. The plugging of the tubes that are adjacent to the retainer bars was performed. MHI has recommended to the purchaser [SCE] to remove the retainer bars that would have the possibility of vibration with large amplitude or to perform the plugging and stabilizing for the associated tubes. According to MHI, it is the lower resonance frequency of the smaller diameter retainer bars that is susceptible to turbulent two-phase flow exciting the bar into its prime resonance or some harmonic frequency thereof [p10, item 3].14 Whatever, a number of the tubes capturing the retainer bar had sustained abraded wear from interaction with it. These tubes comprised six tubes in U2 and four tubes in U3, with seven tubes in total showing wear greater than the 35% limit of the tube wall thickness for which isolation from service is required by plugging with, as previously noted, an incidence site in one of U2 RSGs having worn through 90% of its wall thickness. I agree with the findings of MHI that the tube wear at the retainer bar localities arises because of random flow induced (not FEI) vibration of the retainer bar itself, it being entirely independent of any tube motion excited from other sources. However, MHI's advice to either plug the local tubes and/or remove the retainer bars at risk raises two issues unique to the retainer bar and its sub-assembly: (i) Plugging of the at-risk tubes is not a satisfactory solution because it is the retainer bar that vibrates via random fluid flow processes at sub FEI critical velocity levels - these are likely to continue in play or, indeed, exacerbate at the proposed U2 restart at 70% power, leading to through-tube abrasion, the detachment of tube fragments, lodging at other unplugged and in-service tube localities, resulting in the so-called 'foreign object' tube wear; (ii) MHI's recommendation that those retainer bars at risk of large-amplitude fluid flow excited vibration should be removed is, of course, dependent upon reliable analysis to identify the at-risk assemblies; and, importantly, and (iii) this restraint system probably also serves to contain the tube bundle geometry during a main line steam break (MSLB) design basis event, so any change or removal of the retaining bar assemblage would require a full safety justification." Westinghouse states, "For most of the straight leg section of the tube, the gap velocities at lower power levels and at 100% power are similar. The recirculating fluid flow rate is relatively constant at all power levels. However, in the U-bend region, the gap velocities are a strong function of power level. The steam flow in the bundle is cumulative and increases as a function of the power level and the bundle height which causes high fluid quality, void fraction, and secondary fluid velocities in the upper bundle." SCE in its November 30, 2012, NRC Presentation stated, "Four tubes with retainer bars wear above 35% limit in Unit 2 were plugged." The NRC website states, "The severity of one of the wear indications at a Unit 2 retainer bar was significant enough (90 percent thru-wall) to warrant in-situ pressure testing. This pressure test confirmed the structural integrity of this tube (there was no leakage)." DAB Safety Team Comments: Let us summarize what John Large and Westinghouse are saying: (1) Plugging of the at-risk tubes is not a satisfactory solution because it is the retainer bar that vibrates via random fluid flow processes at sub FEI critical velocity levels - these are likely to continue to vibrate or, indeed, exacerbate at the proposed U2 restart at 70% power, leading to through-tube abrasion, the detachment of tube fragments, lodging at other unplugged and/or in-service tube localities, resulting in the so-called 'foreign object' tube wear, (2) For most of the straight leg section of the tube, the gap velocities at lower power levels and at 100% power are similar. Therefore, even at 70% power, the tube-to-retainer bar wear will continue at the same rate as 100% power and plugging the tubes is not a satisfactory solution in terms of reducing the active tubes rupture safety risks. SCE is not stating the facts either in its Root Cause Evaluation nor in its NRC Presentation. Two better questions are, "How many tubes in Unit 2 have what amounts of fatigue cracks and why has SCE not used state-of-the-art technology to visually examine all RSG tubes at San Onofre?" What this really means is that Southern Californians were lucky once again, that Unit 2 just happened to be shutdown for refueling! Otherwise, one or more worn tubes could have leaked or failed due to a design bases accident and/or any unanticipated transients. Almost 180 tubes had to be plugged and stabilized in Unit 2 Replacement Steam Generators due to retainer bar design mistakes. In addition, no reports are available to determine the extent of tube fatigue damage or damage to the small retainer bars caused by the worn tubes and whether the damaged retaining bars are strong enough to restrain the movement of the anti-vibration bar assembly during a main steam line break design basis event (Ref: NRR RAI #32). The design of the retainer bars approved by SCE and manufactured by MHI clearly violated the Code of Federal Regulations, 10 CFR Part 50, GDC 14, "RCPB—shall have "an extremely low probability of abnormal leakage…and gross rupture" and Appendix B, Criterion III, "Design Control." The DAB Safety Team's opinion is that NRC AIT is treating the retainer bar mistakes and its design approval by SCE just as a routine matter like "No big deal, nothing happened, so who cares" instead of performing the strict enforcement required of an Independent Regulator tasked with ensuring Public Safety. This investigation by the AIT does not meet the Honorable and Respected NRC Chairman's Standards. 3. NRC AIT in its report dated November 11, 2012 (Re: Unresolved Item 05000362/2012007-08, "Non-Conservative Thermal-Hydraulic Model Results") states that, "The licensee and Mitsubishi continued to evaluate this unresolved item and no final conclusions were reached at the time of the inspection. The NRC is continuing to perform independent reviews of existing information, and will conduct additional reviews as new information becomes available." In the original Report in July 2012, the NRC AIT concluded that, "Due to modeling errors, the SONGS replacement generators were not designed with adequate thermal hydraulic margin to preclude the onset of fluid-elastic instability." John Large states, "I identify a number of issues with the … AREVA Tube-to-Tube Report, including: (i) it is not exactly clear which properties are being represented on the spider diagram for comparison with the other operational SGs; even so (ii) since it is most unlikely that AREVA has undertaken a comprehensive (ATHOS) simulation of each of the five nominated SGs, the comparisons drawn are likely to be between aggregate or bulk flows within the entire tube bundle of each SG; (iii) as acknowledged by AREVA, the SONGS RSGs are dominated by in-plane flow regimes whereas all other SGs are characterized by out-of-plane flow regimes; and (iv) none of the comparative SGs has been identified. In other words, … I cannot reason how, are making a direct comparison of the complex two-phase fluid cross-flow situation in the SONGS and other five comparative plant steam generators, then these figures only provide the bases of a somewhat meaningless comparisons. A complete understanding of the causation of the in-plane FEI is essential to ensure that the SONGS Unit 2 plant is acceptably safe to restart and, once restarted, predictably safe to continue in operation over the proposed 150 day inspection interval. To the contrary, the understanding presented by SCE is neither comprehensive nor convincing. In my opinion, simply sweeping the FEI issue under the carpet on the basis of (in- or out-of-plane) FEI will not reoccur at 70% power is not only disingenuous but foolhardy." Arnie Gundersen states, "The AIT report indicated that the change to the FIT-III evaluation methodology was not discussed as part of Edison's 50.59 screening because the details of thermal hydraulic models used for the design of the OSG were not discussed in the original FSAR. It should have been obvious to Edison that FIT-III has not been benchmarked and had not been previously used in licensing procedures showing that the use of FIT-III might have an adverse effect on the FSAR safety analysis thus necessitating the entire license amendment review and public hearing process. As noted by the AIT, Edison approved the use of FIT-III code even though the code was not benchmarked nor identified as acceptable in the FSAR. Consequently, Edison operated San Onofre without knowing the uncertainties in the Replacement Steam Generators' performance characteristics. Predicted liquid levels, pressure drops, vibrations, and temperatures at both Units 2 and 3 were all subject to unknown uncertainties during both normal and abnormal operations. In my opinion, by approving the use of an un-benchmarked and untested design tool like FIT-III, Edison did not meet the requirements expected from a nuclear licensee. Use of an un-benchmarked computer code that is not included in the FSAR protocol demands a formal FSAR license amendment process including the requisite public hearings." Arnie Gundersen further states, "The AIT reported that FIT-III predictions differed considerably in comparison to an Electric Power Research Institute developed code named ATHOS. FIT-III predicted lower flow velocities and void fractions that were not conservative compared to ATHOS. The AIT Report neglected an analysis of the root cause of the critical differences between FIT-III and ATHOS, and the negative impact such lax calculational modeling had on the design, fabrication, and successful operation of the San Onofre RSGs. Had Edison sought the required FSAR license amendment, differences between FIT-III and ATHOS would have been identified six years ago. The AIT did not address the possibility that the lack of conservatism in FIT-III predictions, in addition to causing tube vibrations, could also result in non-conservative predictions of the behavior of the steam generator pressure vessel and associated main steam piping during accident conditions that are required to be analyzed in the FSAR. The AIT noted that the non-conservatisms in FIT-III are a contributor to the failure by Edison to adequately calculate the San Onofre RSG tube vibrations. But equally important, the AIT failed to address that FIT-III could also create non-conservative predictions of the behavior of the steam generator pressure vessel and associated main steam piping during accident conditions that are required to be analyzed in the FSAR. Such a conclusion implies that damage to the steam generator pressure vessel itself, and not just the tubes, might have occurred at San Onofre and remains unanalyzed by either Edison or the NRC. The probability of an accident exceeding the plant's Current Design Basis is increased by the radically different Edison Replacement Steam Generators. Hence, the risks involved in operating the San Onofre RSGs should have been addressed as part of an FSAR license amendment and hearing process. It is my professional opinion that Edison should have applied for the 50.59 process so that the FSAR license amendment evaluation and public hearings would have occurred six years ago, prior to creating an accident scenario and facing losses that by the end of this process will easily total more than $1 Billion. The seriousness of the licensing and safety impact of the damaged RSGs at San Onofre cannot be overstated or underestimated. Any Design Basis Accident (DBA) as defined in the FSAR needs to be accurately modeled in order to protect public health and safety. The FSAR's DBA analysis including the extent of tube leakage in the event of a Main Steam Line Break significantly impacts the design and implementation of Emergency Evacuation Plans. In the event of a steam line break accident in the San Onofre Replacement Steam Generators with the degraded condition of the tubes, an accident would have occurred that is more severe than any design basis accident scenario previously analyzed by Edison in the FSAR. Such a DBA steam line break accident would render the San Onofre emergency plan totally inadequate and most likely cause a permanent evacuation of a large portion of Southern California." DAB Safety Team Comments: After the June 18, 2012 public Meeting, the NRC AIT Team Chief announced to the world, "The computer simulation used by Mitsubishi during the design of the steam generators had under-predicted velocities of steam and water inside the steam generators by factors of three to four times." Now, six months later, the AIT Team is saying the matter is unresolved. The AIT Team is just repeating what SCE says or is not sure what they said four months ago. ATHOS Modeling results are not reliable, because the results by NRC AIT Team, Westinghouse, MHI, AREVA and Independent Experts show that fluid elastic instability occurred both in Units 3 and 2. The investigations in the Root cause of SONGS Unit 3 FEI regarding computer modeling have not been completed by NRC AIT Team, SCE and MHI. FEI did not occur in Unit 2 according to DAB Safety Team and Westinghouse. As also shown in other DAB Safety Team reports, FEI was not caused in Unit 3 by tube-to AVB gaps as claimed by NRC AIT Team and SCE. This is consistent with the findings of Westinghouse, AREVA, MHI, John Large and SONGS Anonymous Insiders. The AIT Team is hurting its own credibility by issuing contradicting and conflicting statements. This investigation by the AIT does not meet the NRC Chairman's Standards. 4. NRC AIT report dated November 11, 2012 (Re: Unresolved Item 05000362/2012007-10, "Evaluation of Departure of Methods of Evaluation for 10 CFR 50.59 Processes") closed the referenced item by stating: (a) The change from ANSYS to ABAQUS did not require a license amendment prior to implementing the change, so with respect to section 2.10.D.6 of the NRC Enforcement Manual, there is no reasonable likelihood that the change from ANSYS to ABAQUS would ever require NRC approval. Therefore, in accordance with the NRC Enforcement Manual, the inspectors determined that the licensee's change from ANSYS to ABAQUS was a minor violation of 10 CFR 50.59(d)(1), and (b)n Based on this, the inspectors determined that the licensee had changed from using ANSYS and STRUDL to analyze several events for the original steam generators, to using only ANSYS to analyze a single limiting event for the replacement steam generators. Therefore, because the licensee did not change the method described in the Updated Final Safety Analysis Report, the inspectors concluded that the licensee did not need to obtain a license amendment prior to implementing that change. In the original Report in July 2012, the NRR technical specialist reviewed SCE's 10 CFR 50.59 evaluation and found two instances that failed to adequately address whether the change involved a departure of the method of evaluation described in the updated final safety analysis report: (a) Use of ABAQUS instead of ANSYS: The SCE's 50.59 evaluation incorrectly determined that using the ABAQUS instead of ANSYS was a change to an element of the method described in the updated final safety analysis report did not constitute changing from a method described in the updated final safety analysis report to another method, and as such, did not mention whether ABAQUS has been approved by the NRC for this application. (b) Use of ANSYS instead of STRUDL and ANSYS: While SCE's 50.59 evaluation correctly considered this a change from a method described in the FSAR to another method, the 50.59 evaluation did not mention whether the method has been approved by NRC for this application. NRC AIT Report states, "For the Unit 2 and Unit 3 replacement steam generators, the licensee determined that the proposed activity did not adversely affect a design function, or the method of performing or controlling a design function described in the updated final safety analysis report. The licensee evaluated the following updated final safety analysis report design functions in the 50.59 screening: Steam Generator Design Functions. Let us examine the effect of these changes on Steam Generator Design Functions: The design functions of the steam generators tubes and tube supports are to: (1.) Limit tube flow-induced vibration to acceptable levels during normal operating conditions, and (2) Prevent a tube rupture concurrent with other accidents. Change Number 1: 105,000 square feet tube heat transfer area in OSGs; 116,100 square feet tube heat transfer area in RSGs; 11.1% increase in heat transfer area, which is more than a minimal change of 10% in the non-conservative direction. Change accomplished by addition of 377 tubes in the central region by removal of stay cylinder and increasing the length of 9727 tubes by > 7 inches in each of the four RSGs. Change Number 2: Operating Secondary Pressure – OSGs: 900 psi, RSG: 833 psi ~ 10% change – A catastrophic change for onset and ongoing exponential fluid elastic instability Change Number 3: Tube wall thickness was reduced from 0.048 inches to 0.043 to pump more reactor coolant through the tubes > 11.6% change - The latest academic research indicates that the tube vibrations become large as T/D decreases and L/D increases, because the in-plane tube vibrations strongly depend on the dynamic characteristics of tubes such as the natural frequency and the damping ability. Four other changes: Moisture content was reduced from 0.2% to 0.1% to improve SG performance, RCS Volume was increased from 1895 cubic feet to 2003 cubic feet, RCS Flow was increased from 198,000 gpm to 209,000 gpm, feedwater flow was increased from 7.4 million pound per hour to 7.6 million pound per hour and AVBs were not designed to prevent against adverse effects of fluid elastic instability (In-plane vibrations, Tube-to-Tube wear, steam dry-outs). These unapproved and unanalyzed changes were claimed to be a conservative decision and improvements in the RSGs from OSGs were presented as a "like for Like" change. No mixing baffles were added in the SONGS RSGs to improve the T/H Performance in the RSGs. FEI and SR Values were not provided by SCE in the RSG Design Specifications. SCE told MHI to avoid the NRC Approval… MHI neither provided in-plane supports, nor provided the operational criteria to prevent FEI in one of the largest steam generators with such high steam flows. MHI did not benchmark CE SG Computer codes or design details, neither did SCE, nor did SCE check the work of MHI. And Honorable and Respected Dr. McFarlane says, "SCE is responsible for the work of its vendors and contractors. Look at Palo Verde RSGs, a Success Story and SONGS RSGs, a $ Billion Blunder…. NRC AIT Report states, "The licensee's bid specification required that the stay cylinder feature of the original steam generators be eliminated to maximize the number of tubes that could be installed in the replacement steam generators and to mitigate past problems with tube wear at tube supports caused by relatively cool water and high flow velocities in the central part of the tube bundle. Mitsubishi employed a broached trefoil tube support plates instead of the egg crate supports in the original design. In addition to providing for better control of tube to support plate gaps and easier assembly, the broached tube support plates were intended to address past problems with the egg crate supports by providing less line of contact and faster flow between the tubes and support plates, reducing the potential for deposit buildup and corrosion." Arnie Gundersen states, "As the NRC confirmed in its AIT report, a large steam void has developed near where the additional tubes were added in the Replacement Steam Generators (called fluid elastic instability) that allows many types of excess vibrations to occur. Fairewinds review of Edison's Condition Report clearly shows that the location within the steam generators where the steam "fluid elastic instability" has developed is precisely the region where the extra heat created by the 400 new tubes would create an excess of steam and various vibrational modes." NRC AIT report states, "Mitsubishi's preliminary explanation of the failure mechanism started with the combination of two factors: (1) a relatively small tube pitch to tube diameter ratio (P/D), and (2) high void fraction in the tube bundle area where the tube-to-tube wear was identified. The small pitch to diameter ratio was a fixed parameter in the replacement steam generators established by the nominal center-to-center distance between adjacent tubes (P) and the nominal outside diameter of the tubes (D). The high void fraction was identified from the results of Mitsubishi's thermal-hydraulic model for the secondary side of the replacement steam generators. Mitsubishi considered that the combination of these two factors may have resulted in favorable conditions for in-plane tube vibration based, in part, on the results of recent studies in fluid-elastic instability." Mitsubishi also states, "Low secondary pressures are severe for vibration." John Large states, "Referring to the short section of the FSAR provided to me by SCE, which I understand is not to be amended for the Unit 2 restart: (a) there is no account of the changes that have been made in the evaluation of the tube structural and leakage integrity, that is from the stage of predicting those tubes at risk of TTW and other forms of wear, the tube thinning wear rates, through to the nature of the tube failure being unique to the type and extent of the wear pattern and tube thinning; and (b) the methods of deducing, mainly by unproven inference, from the probe inspection results particularly to determine the in-plane AVB effectiveness, includes unacceptably large elements of test and experimentation that are inconsistent with the analyses and descriptions of the FSAR." John Large states, "SCE's assertion that reducing power to 70% will at the best alleviate, but not eliminate, the TTW and other modes of tube and component wear is little more than hypothesis - the supporting Operational Assessments and analyses have not proven it to be otherwise. I am of the opinion that trialling this hypothesis by putting the SONGS Unit 2 back into service will, because of the uncertainties and unresolved issues involved, embrace a great deal of change, test and experiment. The terms of the Confirmatory Action Letter of March 11 2012, are versed such that to meet compliance the response of SCE via its Return to Service Report,11 must include considerable changes of conditions and procedures that are outside the reference bounds of the present FSAR – this is because the physical condition of the RSGs, and the means by which this is evaluated and projected into future in-service operation, have substantially and irrevocably changed since the current FSAR was approved. The fact that SCE fails to satisfy the requirements of the CAL is neither here nor there, although it illustrates the scope and complexity of the response required. At the time of preparing the CAL, the NRC being well-versed in the failures at the San Onofre nuclear plant, surely must have known that the only satisfactory response to the CAL would indeed require considerable changes, tests and experiments to be implemented." DAB Safety Team Comments: Therefore, the DAB Safety Team concludes that the changes in design functions of the RSGs tubes and tube supports described above definitely: a) did not limit tube flow-induced vibration to acceptable levels during normal operating conditions and, b) involved a significant reduction in a margin of safety – Failure of 8 Unit 3 SG Tubes under MSLB test conditions and significant TTW > 35% of ~381 tubes in Unit 3 RSGs. A multiple tube failure event, if actually would have occurred during a MSLB would have resulted in a significant increase in the off-site radiological consequences over the single tube burst event, if currently considered in the SONGS approved FSAR by NRC Region IV. The Replacement Steam Generator (RSG) modifications at San Onofre increased both the likelihood of equipment failure and the radiological consequence of such failure and therefore directly affect the FSAR Current Design Basis. The AIT has no business contradicting conclusions made earlier by the NRR technical specialist. This investigation by the AIT does not meet the NRC Chairman's Standards. NRC Region IV Response to DAB Safety Team Analysis of SONGS 10 CFR 50.59 Evaluation Comments: The NRC has already conducted several reviews of the 10 CFR 50.59 documents associated with the replacement of the steam generators at SONGS. These reviews involved NRC inspectors from multiple offices including Region IV, Region II and the Office of Nuclear Reactor Regulation at NRC headquarters. The results of these reviews are contained in NRC two inspection reports that are available at http://www.nrc.gov/info-finder/reactor/songs/tube-degradation.html. [see the Augmented Inspection Team Report dated July 18, 2012, and the Augmented Inspection Team Follow-Up Report dated November 9, 2012]. It is worthy of note that the NRC staff is currently reviewing 10 CFR 50.59 documents associated with the licensee's proposed restart activities. The results of the ongoing review will be documented in a future inspection report. Comments from Mel Silberberg [NRC-RES, Retired [Chief, Severe Accident Research Branch; Waste Management Branch] to Region IV: I am disappointed in the composition of the special panel! Where is the representation from NRC-RES? The issues at SONGS involve thermal hydraulics and material science. The NRC-RES and its contractors are experts in these areas. The Office of Research was created by the Congress for such situations. Two RES staff covering these disciplines and one or two consultants, serving as peer-reviewers. Perhaps there needs to be a separate peer review. Public confidence can only be gained using logical, informed measures as I described above. Inspection Reports are only one facet of the problem, no question. However, understanding the reasons for the fluid instability, possible cavitation corrosion effects, etc. are phenomena which require evaluation by T/H as well as materials experts, with appropriate oversight by the ACRS. The SCE, the nuclear industry, the NRC and the public need assurance, not educated guesses. I have not seen a bona fide attempt to understand resolve the issue such that all can be alert to potential problems. I still remain puzzled as to why the ACRS [at least one of the Subcommittees]. I am trying to reach the ACRS Exec. Director to discuss this point. Thank you." DAB Safety Team Further Comments: NRC Region IV Inspectors need to be re-trained in interpretation of significance of 10 CFR 50.59 Evaluation rules and meaning of changes in design function on safety evaluations. Simply sweeping the 10 CFR 50.59 mistakes under the carpet on the basis of meaningless statements, "The NRC has already conducted several reviews of the 10 CFR 50.59 documents associated with the replacement of the steam generators at SONGS. The present SONGS NRC approved for the total S/G tube leakage assumes a limit of 1 gpm for all S/Gs, which ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10CFR100 limits in the event of either a S/G tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm (720 gpd) leakage limit per S/G ensures that S/G tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions." These reviews of 10 CFR 50.59 and SONGS FSAR S/G tube rupture limits from NRC inspectors from multiple offices including Region IV, Region II and the Office of Nuclear Reactor Regulation at NRC headquarters" are not only disingenuous but foolhardy. A single tube leakage and/or rupture could result in a nuclear incident or accident with tube leakages assumed in the current SONGS FSAR as shown in the Table below. A multiple tube failure event for all phases of the reactor in-core fuel cycle, would result in a significant increase in the off-site radiological consequences (e.g., Fukushima, Chernobyl, etc.) over the single tube burst event currently considered in the FSAR. The rapid and extraordinarily severe wear that resulted in the 2012 failures of all of Edison's San Onofre Replacement Steam Generators was the result of Edison's 2005 decision to radically change the RSG design and to claim that the Part 50.59 licensing process did not apply. Arnie Gundersen and DAB Safety Team have stated consistently that San Onofre Replacement Steam Generator tube damage discovered in 2012 was so severe and extensive that both reactors have been operating in violation of their NRC FSAR license design basis as defined in their Technical Specifications. While the NRC Augmented Inspection Team (AIT) briefly described how Edison addressed its 50.59 requirements, the evidence shows that Edison did not comply with the NEI guidelines for implementing 50.59. Published reports indicate that the strategic decision made by Edison that the 50.59 process would not be applied to the RSGs was made by corporate officials before any engineering personnel had actually performed the 50.59 engineering analysis. Consequently, Edison made a management decision to claim that the 50.59 process did not apply and therefore San Onofre was not required to seek NRC approval for the proposed changes at San Onofre Units 2 and 3. These unlicensed unapproved design changes to the containment boundary violated Federal Regulations and therefore the FSAR must be amended prior Unit 2 Restart to reflect multiple steam generator tube ruptures with MSLB plus DBE due to Edison's significant untested and unanalyzed modifications.The DAB Safety Team: Don, Ace and a BATTERY of safety-conscious San Onofre insiders plus industry experts from around the world who wish to remain anonymous. These volunteers assist the DAB Safety Team by sharing knowledge, opinions and insight but are not responsible for the contents of the DAB Safety Team's reports. We continue to work together as a Safety Team to prepare additional DAB Safety Team Documents, which explain in detail why a SONGS restart is unsafe at any power level without a Full/Thorough/Transparent NRC 50.90 License Amendment and Evidentiary Public Hearings. Our Mission: To prevent a Trillion Dollar Eco-Disaster like Fukushima, from happening in the USA. Copyright January 31, 2013 by The DAB Safety Team. All rights reserved. This material may not be published, broadcast or redistributed without crediting the DAB Safety Team. The contents cannot be altered without the Written Permission of the DAB Safety Team Leader and/or the DAB Safety Team's Attorney... Thanks to NRC Moderator for Posting this Blog. HAHN Baba

comment #46729 posted on 2013-01-25 12:40:45 by CaptD in response to comment #45347

Edwin Hackett, Executive Director ACRS ==> Main number is 301-415-7360 Thanks for your reply, and staying aware of what is happening at SONGS aka SanO.

comment #46781 posted on 2013-01-25 15:07:08 by HelpAllHurtNeverBaba

Portions of the following information has been extracted from the DAB Safety Team Reports (Search Google Drive for DAB Safety Team & Related Info). DAB Safety Team is a group of Public Service Oriented Southern Californians and Anonymous San Onofre Insiders trying to help the NRC and Public by providing unbiased, logical and factual information to assess the real dangers of San Onofre Unit 2. Unit 2 permission for restart by NRC is imminent and REAL Root Cause for destruction of $1 Billion Units 2 and 3 RSGs (Includes equipment cost and expenses) has not even been determined. Public Safety by NRC in a rush to judgment cannot be compromised due to please profit-motivated SCE. Commenting on the NRC Augmented Inspection Team San Onofre Report… Just trying to help NRC Augmented Inspection Team Chief and NRC San Onofre Special Panel.. Thanking to the Moderator for posting this comment HAHN Baba NOTE: Highly recommend that NRC Augmented Inspection Team and NRC San Onofre Special Panel thoroughly review SONGS Unit 2 Return to Service MHI, AREVA, Westinghouse, DAB Safety Team and John Large Reports and carefully examine the operational differences between Unit 2 and 3 and then update the NRC AIT report with real Root cause for FEI in Unit 3 and NO FEI in Unit 2. The AIT inspection concluded that: (1) SCE was adequately pursuing the causes of the unexpected steam generator tube-to-tube degradation. In an effort to identify the causes, SCE retained a significant number of outside industry experts, consultants, and steam generator manufacturers, including Westinghouse and AREVA to perform thermal-hydraulic and flow induced vibration modeling and analysis; (2) The combination of unpredicted, adverse thermal hydraulic conditions and insufficient contact forces in the upper tube bundle caused a phenomenon called “fluid-elastic instability” which was a significant contributor to the tube to tube wear resulting in the tube leak. The team concluded that the differences in severity of the tube-to-tube wear between Unit 2 and Unit 3 may be related to the changes to the manufacturing/fabrication of the tubes and other components which may have resulted in increased clearance between the anti-vibration bars and the tubes; (3) Due to modeling errors, the SONGS replacement generators were not designed with adequate thermal hydraulic margin to preclude the onset of fluid-elastic instability. Unless changes are made to the operation or configuration of the steam generators, high fluid velocities and high void fractions in localized regions in the u-bend will continue to cause excessive tube wear and accelerated wear that could result in tube leakage and/or tube rupture; (4) The thermal hydraulic phenomena contributing to the fluid-elastic instability is present in both Unit 2 and 3 steam generators; (5) Based on the updated final safety analysis report description of the original steam generators, the steam generators major design changes were appropriately reviewed in accordance with the 10 CFR 50.59 requirements. So based on a review of the AIT Report and World’s Experts, the potential causes, which were significant contributors to the “fluid-elastic instability” in SONGS Unit 3 and the tube-to-tube wear resulting in the tube leak are as follows: A. Insufficient contact tube-to AVB forces and differences in manufacturing/fabrication of the tubes and other components between Units 2 & 3 B. Due to modeling errors, the SONGS replacement generators were not designed with adequate thermal hydraulic margin to preclude the onset of fluid-elastic instability. C. Operational Factors A. Let us now examine that whether insufficient contact tube-to AVB forces in the Unit 3 upper tube bundle caused “fluid-elastic instability” which was a significant contributor to the tube-to-tube wear resulting in the tube leak. A.1- MHI states, “By design, U-bend support in the in-plane direction was not provided for the SONGS SG’s”. In the design stage, MHI considered that the tube U-bend support in the out-of-plane direction designed for “zero” tube-to-AVB gap in hot condition was sufficient to prevent the tube from becoming fluid-elastic unstable during operation based on the MHI experiences and contemporary practice. MHI postulated that a “zero” gap in the hot condition does not necessarily ensure that the support is active and that contact force between the tube and the AVB is required for the support to be considered active. The most likely cause of the observed tube-to-tube wear is multiple consecutive AVB supports becoming inactive during operation. This is attributed to redistribution of the tube-to-AVB-gaps under the fluid hydrodynamic pressure exerted on the tubes during operation. This phenomenon is called by MHI, “tube bundle flowering” and is postulated to result in a spreading of the tube U-bends in the out-of-plane direction to varying degrees based on their location in the tube bundle (the hydrodynamic pressure varies within the U bend). This tube U-bend spreading causes an increase of the tube-to-AVB gap sizes and decrease of tube-to-AVB contact forces rendering the AVB supports inactive and potentially significantly contributing to tube FEI. Observations Common to BOTH Unit-2 and Unit-3: The AVBs, end caps, and retainer bars were manufactured according to the design. It was confirmed that there were no significant gaps between the AVBs and tubes, which might have contributed to excessive tube vibration because the AVBs appear to be virtually in contact with tubes. MHI states, “The higher than typical void fraction is a result of a very large and tightly packed tube bundle, particularly in the U-bend, with high heat flux in the hot leg side. Because this high void fraction is a potentially major cause of the tube FEI, and consequently unexpected tube wear (as it affects both the flow velocity and the damping factors).” A.2 - AREVA states, “At 100% power, the thermal-hydraulic conditions in the U-bend region of the SONGS replacement steam generators exceeded the past successful operational envelope for U-bend nuclear steam generators based on presently available data. The primary source of tube-to-AVB contact forces is the restraint provided by the retaining bars and bridges, reacting against the component dimensional dispersion of the tubes and AVBs. Contact forces are available for both cold and hot conditions. Contact forces significantly increase at normal operating temperature and pressure due to diametric expansion of the tubes and thermal growth of the AVBs. After fluid elastic instability develops, the amplitude of in-plane motion continuously increases and the forces needed to prevent in-plane motion at any given AVB location become relatively large. Hence shortly after instability occurs, U-bends begin to swing in Mode 1 and overcome hindrance at any AVB location.” A.3 - Westinghouse states, “Test data shows that the onset of in-plane (IP) vibration requires much higher velocities than the onset of out-of-plane (OP) fluid-elastic excitation. Hence, a tube that may vibrate in-plane (IP) would definitely be unstable OP. A small AVB gap that would be considered active in the OP mode would also be active in the IP mode because the small gap will prevent significant in-plane motion due to lack of clearance (gap) for the combined OP and IP motions. Thus, a contact force is not required to prevent significant IP motion. Manufacturing Considerations: There are several potential manufacturing considerations associated with review of the design drawings based on Westinghouse experience. The first two are related to increased proximity potential that is likely associated with the ECT evidence for proximity. Two others are associated with the AVB configuration and the additional orthogonal support structure that can interact with the first two during manufacturing. Another relates to AVB fabrication tolerances. These potential issues include: (1) The smaller nominal in-plane spacing between large radius U-bend tubes than comparable Westinghouse experience, (2) The much larger relative shrinkage of two sides (cold leg and hot leg) of each tube that can occur within the tubesheet drilling tolerances. Differences in axial shrinkage of tube legs can change the shape of the U-bends and reduce in-plane clearances between tubes from what was installed prior to hydraulic expansion, (3) The potential for the ends of the lateral sets of AVBs (designated as side narrow and side wide on the Design Anti-Vibration Bar Assembly Drawing that are attached to the AVB support structure on the sides of the tube bundle to become displaced from their intended positions during lower shell assembly rotation, (4) The potential for the 13 orthogonal bridge structure segments that are welded to the ends of AVB end cap extensions to produce reactions inside the bundle due to weld shrinkage and added weight during bundle rotation, and (5) Control of AVB fabrication tolerances sufficient to avoid undesirable interactions within the bundle. If AVBs are not flat with no twist in the unrestrained state they can tend to spread tube columns and introduce unexpected gaps greater than nominal inside the bundle away from the fixed weld spacing. The weight of the additional support structure after installation could accentuate any of the above potential issues. There is insufficient evidence to conclude that any of the listed potential issues are directly responsible for the unexpected tube wear, but these issues could all lead to unexpected tube/AVB fit-up conditions that would support the amplitude limited fluid-elastic vibration mechanism. None were extensively treated in the SCE root cause evaluation." A.4 -HAHN Baba concludes that SONGS Unit 3 RSG's were operating outside SONGS Technical Specification Limits for Reactor Thermal Power and Current Licensing Basis for Design Basis Accident Conditions. HAHN Baba further agrees with MHI that high steam flows and cross-flow velocities combined with narrow tube pitch-to-diameter ratio caused elastic deformation of the U-tube bundle from the beginning of the Unit 3 cycle, which initiated the process of tube-to-AVB wear and insufficient contact forces between tubes and AVBs. Tube bundle distortion is considered a major contributing cause to the mechanism of tube-to-tube/AVB/TSP wear seen in the Unit 3 SG's. After 11 months of wear, contact forces were virtually eliminated between the tube and AVBs in the areas of highest area of Unit 3 wear as confirmed by ECT and visual inspections. Therefore, based on a review of MHI, AREVA and Westinghouse excerpts shown below, the HAHN Baba concludes that FEI and MHI Flowering effect redistributed the tube-to-AVB gaps in Unit 3 RSG's. It is the HAHN Baba's opinion that NRC and SCE claims that insufficient contact forces in Unit 3 Tube-to-AVB Gaps ALONE caused tube "to" tube wear are misleading, erroneous and designed to put the blame on MHI for purposes of making SCE look good in the public's eyes and for collecting insurance money from MHI's manufacturing so called defects. B. Let us now examine of effects of modeling errors, that the SONGS replacement generators were not designed with adequate thermal hydraulic margin to preclude the onset of fluid-elastic instability. B.1 - NRC AIT Report states, "The ATHOS thermal-hydraulic model predicts bulk fluid behavior based on first principals and empirical correlations and as a result, it is not able to evaluate mechanical, fabrication, or structural material differences or other phenomena that may be unique to each steam generator. Therefore this analysis cannot account for these mechanical factors and differences which could very likely also be contributing to the tube degradation." B.2 - Ivan Cotton states, "Fluid elastic instability is one of the most damaging types of instabilities encountered in heat exchangers and steam generators and can impose a severe economic penalty on the power and chemical industries. At present our understanding of the mechanisms leading to fluid-elastic instability is very limited and more experiments are needed to more fully delineate the conditions for the onset of fluid-elastic instability." Such experimentation should only be done in a sealed lab, NOT our environment with the lives of eight million local residents at stake in the outcome! B.3 - Ishihara, Kunihiko and Kitayama state, "Tube vibrations become large as tube thickness/diameter ratio (T/D) increases and tube length/diameter ratio (L/D) decreases, and the tube vibrations strongly depend on the dynamic characteristics of tubes such as the natural frequency and the damping ability." B.4 - Fairewinde states, "Realistically, the 3-D steam analysis is not accurate enough to apply to such important safety related determinations. To make such mathematical risk 3-D analysis, a very large margin of error must be applied, and that has not been done. For example, if the 3-D steam analysis determines that plugging 100 tubes is a solution, then plugging ten times that number might be the appropriate solution due to the mathematical errors in the 3-D analysis being applied by Edison and Mitsubishi." B.5 - Mitra, V.K. Dhir, I. Catton state, " Flow induced vibrations in heat exchanger tubes have led to numerous accidents and economic losses in the past. Efforts have been made to systematically study the cause of these vibrations and develop remedial design criteria for their avoidance. Instability was clearly seen in single phase and two-phase flow and the critical flow velocity was found to be proportional to tube mass. It is also found that nucleate boiling on the tube surface is also found to have a stabilizing effect on fluid-elastic instability. B.6 - SCE states that SONGS Unit 3 Damage (FEI) was caused due to outdated MHI Thermal-Hydraulic Computer Models. According to NRC AIT Report, SONGS did not specify the value of FEI in its Design and Performance Specifications SO23-617-1. Academic Researchers have discussed and warned about the adverse effects of fluid elastic instability (tube-to-tube wear) in nuclear steam generators since 1970's. Westinghouse and Combustion Engineering (CE) have designed CE engineering replacement steam generators (RSGs) to prevent the adverse effects of fluid elastic instability since 2000's (e.g., PVNGS). B.7 - The NRC AIT Report dated November 9, 2012 states, "the FIT-III thermal-hydraulic model was still in-progress at the time of the inspection and no final conclusions were reached for the cause of the non-conservative flow velocities, which were used as inputs in the tube vibration analysis and resulted in non-conservative stability ratios. Since the licensee had not completed the cause evaluation for this unresolved item, the inspectors were not able to make a final determination of whether a performance deficiency or violation of NRC requirements occurred. The inspectors were informed that Mitsubishi was performing an evaluation of the potential factors that contributed to the low flow velocities in FIT-III relative to the velocities calculated by the ATHOS model developed after the tube leak event in Unit 3. This evaluation was included in Document SO23-617-1-M1530, Revision 1, which also intended to demonstrate the validity of FIT-III results for the original tube vibration analysis. This evaluation was still being finalized and not yet approved by Edison. The licensee and Mitsubishi continued to evaluate this unresolved item and no final conclusions were reached at the time of the inspection. The NRC is continuing to perform independent reviews of existing information, and will conduct additional reviews as new information becomes available. In another related finding, NRC inspectors stated, "SCE Engineers did not meet Procedure SO123-XXIV-37.8.26 requirements to ensure the design of the retainer bar was adequate with respect to the certified design specification. Specifically, the licensee failed to ensure that there was sufficient analytical effort in the design methodology of the anti-vibration bar assembly to support the conclusion that tube wear would not occur as a result of contact with the retainer bars due to flow-induced vibration. The inspectors determined that the requirements for flow-induced vibration in the certified design specification, along with the expectations in Procedure SO123-XXIV-37.8.26, provided sufficient information to reasonably foresee the inadequate design of the retainer bars during the review and approval of design Calculations SO23-617-1-C749 and SO23-617-1-C157, including the associated design drawings provided by Mitsubishi. B.8 – Arnie Gundersen states, "Not only is Mitsubishi unfamiliar with the tightly packed CE design, but Edison's engineers added so many untested variables to the new fabrication that this new design had a significantly increased risk of failure. As a result of the very tight pitch to diameter ratios used in the original CE steam generators, Mitsubishi fabricated a broached plate design that allows almost no water to reach the top of the steam generator. The maximum quality of the water/steam mixture at the top of the steam generator in the U-Bend region should be approximately 40 to 50 percent, i.e. half water and half steam. With the Mitsubishi design the top of the U-tubes are almost dry in some regions. Without liquid in the mixture, there is no damping against vibration, and therefore a severe fluid-elastic instability developed. Because of the Edison/Mitsubishi steam generator changes, the top of the new steam generator is starved for water therefore making tube vibration inevitable. Furthermore, the problem appears to be exacerbated by Mitsubishi's three-dimensional thermal-hydraulic analysis determining how the steam and water mix at the top of the tubes that has been benchmarked against the Westinghouse design but not the original CE design. The real problem in the replacement steam generators at San Onofre is that too much steam and too little water is causing the tubes to vibrate violently in the U-bend region. The tubes are quickly wearing themselves thin enough to completely fail pressure testing. Even if the new tubes are actively not leaking or have not ruptured, the tubes in the Mitsubishi fabrication are at risk of bursting in a main steam line accident scenario and spewing radiation into the air." B.9 - John Large, Internationally Known Scientist and Chartered Nuclear Engineer from London says about the SONGS Unit 2 Replacement Steam Generators (RSGs) AVB Structure, "It impossible to reliably predict the effectiveness of the many thousands of AVB contact points for when the tube bundle is in a hot, pressurized operational state. The combination of the omission of the in-plane AVB restraints, the unique in-plane activity levels of the SONGS RSGs, together the very demanding interpretation of the remote probe data from the cold and depressurized tube inspection, render forecasting the wear of the tubes and many thousands of restraint components when in hot and pressurized service very challenging indeed." B.10 - HAHN Baba Comment to Limitations of ATHOS thermal-hydraulic Models: SCE and MHI are both negligent because they did a very poor job of Industry and Academic Research benchmarking regarding the applicability of thermal-hydraulic computer models during the redesign of SONGS original CE SGs. SCE is negligent because they did not check the results of MHI's outdated Thermal-Hydraulic Computer Models to meet their specification requirements. This does not meet the NRC Chairman's Standards. Therefore, the DAB Safety Team concludes that SCE claims as stated above are not factual. SCE engineers did not check the work of MHI with a critical and questioning attitude and did not meet the 10CFR50, Appendix B, Quality assurance Standards and or NRC Regulations C. Let us now examine the other operational factors, which were significant contributors to the "fluid-elastic instability" in SONGS Unit 3 and the tube-to-tube wear resulting in the tube leak. C.1 - Low steam generator pressures (1 causes the onset of FEI). At the onset of FEI, U-tube bundle tubes start vibrating with very large amplitudes in the in-plane directions. Extremely hot and vibrating tubes need a little amount of water (aka damping, 1.5% water, steam-water mixture vapor Fraction 99.5%). When this happens, the extremely hot and vibrating tubes cannot dissipate their energy and return to their original in-plane design position. In effect, one unstable tube drives its neighbor to instability through repeated violent and turbulent impact events which causes tube leakage, tube failures at MSLB test conditions and or unprecedented tube-tube wear, Tube-to-AVB/Tube Support Plates wear, as we saw in SONGS Unit 3. So in review, due to narrow tube pitch to tube diameter, low tube wall thickness/diameter ratio, high tube length/diameter ratio, low tube clearences, in certain portions of the RSGs U-tubes bundle, fluid velocities exceed the critical velocities due to extremely high steam flows (100% SONGS power conditions outside the industry NORM). These high fluid velocities cause U-tubes to vibrate with very large amplitudes in the in-plane direction and literally hit other the tubes with repeated and violent impacts. Due to lower steam operating pressures (required to generate more heat, electricity and profits) and excessive pressure drops due to high flows and velocities, steam saturation temperature drops. This lowering of steam temperature combined with high heat flux in the hot leg side of the U-tube bundle causes steam dry-outs to form (Vapor fraction >99%), known as "NO Effective Thin Tube Film Damping." Thin film damping refers to the tendency of the steam inside the generators to create a thin film of water between the RSG tubes and the support structures. That film is enough to help keep the tubes from vibrating with large amplitudes, hitting other tubes violently, and protect the Anti-Vibration Bar support structures and maintain the tube-to-AVB gaps and contact forces. These adverse conditions in SONGS at 70% power operation (RTP) with the present defective design and degraded of RSGs known as fluid elastic instability (Tube-to-Tube Wear, or TTW) can lead to rapid U-tubes failure from fatigue or tube-to-tube wear in Unit 2 due to a main steam line break as seen in SONGS Unit 3 RSG's. In summary, FEI is a phenomenon where due to SONGS RSGs design intended for high steam flows causes the tubes to vibrate with increasingly larger amplitudes due to the fluid effective flow velocity exceeding its specific limit (critical velocity) for a given tube and its supporting conditions and a given thermal hydraulic environment. This occurs when the amount of energy imparted on the tube by the fluid is greater than the amount of energy that the tube can dissipate back to the fluid and to the supports. Nucleate boiling on the tube surface or a little amount of water (aka damping, 1.5% water, steam-water mixture vapor fraction <98.5%) is found to have a stabilizing effect on fluid-elastic instability. C.2 – For more information, please see comments posted by HelpAllHurtNeverBaba, January 18, 2013 at 12:20 am on this blog

comment #46276 posted on 2013-01-24 12:12:33 by CaptD in response to comment #45861

@ Mr. Dricks "No, the Advisory Committee on Reactor Safety (ACRS) has not requested a meeting with the NRC technical staff on SONGS related issues. Why not? Especially since SanO's RSG tubing now has more damage that ALL the rest of the nuclear fleet combined! What are they waiting for, and how would a public person contact the Chief of ACRS?

comment #46275 posted on 2013-01-24 12:12:04 by Moderator in response to comment #45861

Publicly available NRC documents related to the steam generator problems at San Onofre are posted in ADAMS and on the special web page at: http://www.nrc.gov/info-finder/reactor/songs/tube-degradation.html

comment #46267 posted on 2013-01-24 12:05:44 by CaptD in response to comment #45347

RE: Mel Silberberg, please look at any of the end of any of DAB Safety Team's documents for more about DAB... https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit

comment #45323 posted on 2013-01-21 16:56:19 by in response to comment #43323

The issues involved in the SONGS steam generator encompass thermal-hydraulics and material science and technology. I am extremely upset and disappointed in the lack of judgement displayed by senior NRC management and the Commission in the glaring omission of the NRC Office of Research[RES] from playing a major role in this special panel. It is at times like this that RES was created by the Congress.to get an independent, confirmatory assessment of abnormal behavior in a nuclear plant.hThe Chairman should insist on the following additions to the panelOne staff expert on thermal-hydraulicsand one staff expert in material science. In addition two consultants from the unverities and or national labs serve on the panel, as peer reviewers. you can not win public confidence in your findingwithout these additions to the to the panel. Mel Silberberg, NRC RES Retired; former Chief of Severe Accident Research Branch.

comment #45344 posted on 2013-01-21 18:31:41 by Mel Silberberg

I am disappointed in the composition of the special panel! Where is the representation from NRC-RES? The issues at SONGS involve thermal hydraulics and material science. The NRC-RES and its contractors are experts in these areas. The Office of Research was created by the Congress for such situations. Two RES staff covering these disciplines and one or two consultants, serving as peer-reviewers. Perhaps there needs to be a separate peer review. Public confidence can only be gained using logical, informed measures as I described above. Mel Silberberg, NRC-RES, Retired [Chief, Severe Accident Research Branch; Waste Management Branch.

comment #46249 posted on 2013-01-24 11:29:36 by Moderator in response to comment #45927

Reasonable assurance is given when licensees comply with NRC regulations. That said, the NRC is always looking at the adequacy of its regulations to ensure safety. Victor Dricks

comment #46250 posted on 2013-01-24 11:30:56 by Moderator in response to comment #45861

No, the Advisory Committee on Reactor Safety (ACRS) has not requested a meeting with the NRC technical staff on SONGS related issues. Victor Dricks

comment #45347 posted on 2013-01-21 18:37:45 by Mel Silberberg in response to comment #45344

correction--on the fifth line, after 'peer-reviewers' please add [should be added to the panel.]

comment #46245 posted on 2013-01-24 11:25:17 by CaptD

In reply to Mr. Silberberg: You sir are correct, we need MORE not LESS information made public in order that knowledgeable people can fact check exactly what is happening at SanO. To hide most of the data behind a veil of secrecy, is no longer acceptable especially since that practice is what has resulted in the current 1 to 1.5 billion dollar debacle at SanO. This is the first time in the US Nuclear Fleet that what Dr. Joram Hopenfeld, (who also retired from the NRC staff) first described (what we now call the Hopenfeld Effect) as a cascade of SG tube failures, has actually been observed in a Steam Generator (See Response to NRR RAI -32 - Technical ==> Attachment 3 https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=0BweZ3c0aFXcFX05DMWxKNmZXUTA). snip "The concerns raised by Dr. Hopenfeld are extremely important safety issues. As the ACRS stated: • Steam generators constitute more than 50% of the surface area of the primary pressure boundary in a pressurized water reactor. • Unlike other parts of the reactor pressure boundary, the barrier to fission product release provided by the steam generator tubes is not reinforced by the reactor containment as an additional barrier." • Leakage of primary coolant through openings in the steam generator tubes could deplete the inventory of water available for the long-term cooling of the core in the event of an accident. In the decade since Dr. Hopenfeld first raised his safety concerns, the NRC has allowed many nuclear plants to continue operating nuclear power plants with literally thousands of steam generator tubes that are known to be fatigue cracked! The ACRS concluded that the NRC staff made these regulatory decisions using incomplete and inaccurate information. After receiving the ACRS's report, the NRC staff considered Hopenfeld's concerns "resolved" even though it had taken no action to address the numerous recommendations in the ACRS report. The NRC must now formally address Dr. Hopenfeld's concerns as soon as possible. In the interim, the NRC must stop making decisions affecting the lives of millions of Americans when it lacks "defensible technical basis" because the US cannot afford a Trillion Dollar Eco-Disaster like Fukushima, due to RSG tube failures caused by poor design, fatigue or any other combination of reasons." Because the Hopenfeld Effect has now been proven as factual, the NRC must re-evaluated it's "dated" thinking and its computer modeling about SG failures which now only allows for a single SG tube failure ASAP... In fact, I predict that time will show that a nuclear accident (not a nuclear incident) was narrowly avoided at SanO on January 31, 2012 only because of shear luck, due to the timing of the discovery of Edison's poorly in-house designed replacement steam generators (RSG). Had that Unit 3 tube been just a tiny bit stronger and not leaked when it did; then with both Unit 2 & 3 back online when a MSLB occurred, we now know that it would have resulted in the complete venting of the core coolant within minutes... This is why what happened at SanO (as the locals like to say) is so important and why the NRC has to "get it right" this time; the safety of the entire US nuclear fleet depends upon it! Just as many basic design problems were discovered after the Fukushima tragedy, Sano has become the model of what NOT to do for all future RSG design engineers globally and demonstrates beyond a shadow of a doubt why having a qualified public review process is so important, especially where the risk of a radioactive "Trillion Dollar Eco-Disaster" is involved.

comment #45365 posted on 2013-01-21 20:35:25 by HelpAllHurtNeverBaba

Trying to help San Onofre Special Panel,,, Thanks to NRC Moderation for Posting... HAHNBaba Credit of DAB Safety Team Press Release 12-12-20 Prior to Issue of any decision regarding restart for Unit 2, SCE needs to demonstrate the viability of Operator Actions for an earthquake, main steam line break or other unanticipated transients in a Full NRC/FEMA Evaluated Emergency Plan Exercise collocated by NRC Head Quarters and evaluated by IPC/Industry Emergency Preparedness/Reactor Oversight and NRR

Experts using the following: • Fully Staffed Control Room or Simulator, Technical Support Center, Operations Support Center, Emergency Offsite Facility, Joint Information Center and Fire Department • Ability for Accurate & Timely detection of a tube leak using N-16 radiation detection system and initiation of operator actions • Ability for Accurate & Timely Identification, Trouble Shooting, Diagnostics and Mitigation of the above events using VLPMS accelerometers for detecting actual tube vibrations for fluid elasticity Mitigation • Ability for Accurate & Timely demonstration of actual tube vibration noise from background noise and the required threshold identification criteria, that would be applied to reach the conclusion that tube vibration is occurring and the number of affected damaged and worn tubes • Ability for Accurate & Timely use of Emergency, Abnormal & Severe Accident Management Procedures • Demonstration of Excellent Communications, Solid Team Work & Alignment, Critical Questioning & Investigative Attitude between all Emergency Operating Facilities, NRC Headquarters, Federal Emergency Management Agency, State of California and Offsite Agencies including Offsite Dose Assessment Committee, California Highway Patrol, Fire Departments, News Media, Emergency Medical Facilities and Public Interest Groups • Ability for demonstration of Accurate & Timely Emergency Declarations, Offsite Notifications / Communications, and Protective Actions Recommendations & Decisions • Ability for prompt notification, evacuation and/or sheltering of disabled, transient and permanent residents in the Emergency Planning Zone during rush traffic hours Acceptance Criteria: • 100 % Accuracy in Emergency Declarations, Offsite Notifications / Communications, and Protective Actions Recommendations & Decisions • No more than 5 Drill/Exercise Weaknesses

comment #49313 posted on 2013-01-30 12:48:18 by HelpAllHurtNeverBaba in response to comment #45347

Hi Mr. Mel Silberberg Can you please comment on the following..... Thanks HAHN Baba Subject: FYI - Exchange of Notes To: Victor.Dricks@nrc.govHelpAllHurtNeverBaba January 29, 2013 at 1:04 am Request for independent re-review of SONGS 50.59 Screen/Evaluation by NRC Region II – Please send me an email after you complete the review ASAP. These guys who performed the screen and evaluations are very close friends of mine and I want to make sure they were on the right track. Trying to help my friends and NRC Region IV. Thanks… HAHN Baba Reply Moderator January 29, 2013 at 2:21 pm The NRC has already conducted several reviews of the 10 CFR 50.59 documents associated with the replacement of the steam generators at SONGS. These reviews involved NRC inspectors from multiple offices including Region IV, Region II and the Office of Nuclear Reactor Regulation at NRC headquarters. The results of these reviews are contained in NRC two inspection reports that are available at http://www.nrc.gov/info-finder/reactor/songs/tube-degradation.html. [see the Augmented Inspection Team Report dated July 18, 2012, and the Augmented Inspection Team Follow-Up Report dated November 9, 2012]. It is worthy of note that the NRC staff is currently reviewing 10 CFR 50.59 documents associated with the licensee's proposed restart activities. The results of the ongoing review will be documented in a future inspection report. Victor Dricks Reply ◦ HelpAllHurtNeverBaba January 29, 2013 at 8:34 pm Your comment is awaiting moderation. Mr. Dricks, Respectfully, Along with Arnie Gundersen and John Large, I totally disagree with the NRC assessments on SONGS 10 CFR 50.59 RSG Evaluations. I was qualified SONGS 50.59 Screener/Evaluator for a decade besides being qualified at several other nuclear power plants. I have performed numerous 50.59 changes and reviews at SONGS. The changes shown below were claimed by Edison to be in the conservative direction and improvements. NRC AIT Report states, "For the Unit 2 and Unit 3 replacement steam generators, the licensee determined that the proposed activity did not adversely affect a design function, or the method of performing or controlling a design function described in the updated final safety analysis report. The licensee evaluated the following updated final safety analysis report design functions in the 50.59 screening: Steam Generator Design Functions…. Let us examine the effect of these changes on Steam Generator Design Functions and then you go back to your peers for more soul searching/research and provide more arguments and we will go from there: The design functions of the steam generators tubes and tube supports are to: (1.) Limit tube flow-induced vibration to acceptable levels during normal operating conditions, and (2) Prevent a tube rupture concurrent with other accidents. Change Number 1: 105,000 square feet tube heat transfer area in OSGs; 116,100 square feet tube heat transfer area in RSGs; 11.1% increase in heat transfer area, which is more than a minimal change of 10% in the non-conservative direction. Change accomplished by addition of 377 tubes in the central region by removal of stay cylinder and increasing the length of 9727 tubes by > 7 inches. Change Number 2: Operating Secondary Pressure – OSGs: 900 psi, RSG: 833 psi ~ 10% change Change Number 3: Tube wall thickness was reduced from 0.048 inches to 0.043 to pump more reactor coolant through the tubes > 10% change Other changes: Moisture content was reduced from 0.2% to 0.1% to improve SG performance, RCS Volume was increased from 1895 cubic feet to 2003 cubic feet, RCS Flow was increased from 198,000 gpm to 209,000 gpm, feedwater flow was increased from 7.4 million pound per hour to 7.6 million pound per hour and AVBs were not designed to prevent against adverse effects of fluid elastic instability (In-plane vibrations, Tube-to-Tube wear, steam dry-outs). These unapproved and unanalyzed changes were claimed to be a conservative decision and improvements in the RSGs from OSGs were presented as a "like for Like" change. No mixing baffles were added in the SONGS RSGs to improve the T/H Performance in the RSGs. FEI and SR Values were not provided by SCE in the RSG Design Specifications. SCE told MHI to avoid the NRC Approval…… MHI did not either provided in-plane supports, or provided the operational criteria to prevent FEI in one of the largest steam generators with such high steam flows. MHI did not benchmark CE SG Computer codes or design details, neither did SCE, nor did SCE check the work of MHI. And Dr. McFarlane says, "SCE is responsible for the work of its vendors and contractors. Look at Palo Verde RSGs, a Success Story and SONGS RSGs, a $ Billion Blunder…. NOTE: ATHOS Modeling results are not reliable, because the results by NRC AIT Team, Westinghouse, MHI, AREVA and Independent Experts show that fluid elastic instability occurred both in Units 3 and 2. The investigations in the Root cause of SONGS Unit 3 FEI regarding computer modeling have not been completed by NRC AIT Team, SCE and MHI. FEI did not occur in Unit 2 according to DAB Safety Team and Westinghouse. As also shown in other DAB Safety Team reports, FEI was not caused in Unit 3 by tube-to AVB gaps as claimed by NRC AIT Team and SCE. This is consistent with the findings of Westinghouse, AREVA, MHI, John Large and SONGS Anonymous Insiders. NRC AIT Report states, "The licensee's bid specification required that the stay cylinder feature of the original steam generators be eliminated to maximize the number of tubes that could be installed in the replacement steam generators and to mitigate past problems with tube wear at tube supports caused by relatively cool water and high flow velocities in the central part of the tube bundle. Mitsubishi employed a broached trefoil tube support plates instead of the egg crate supports in the original design. In addition to providing for better control of tube to support plate gaps and easier assembly, the broached tube support plates were intended to address past problems with the egg crate supports by providing less line of contact and faster flow between the tubes and support plates, reducing the potential for deposit buildup and corrosion." Problems in SONGS Original CE Steam Generators: In the Original 2001 Power Uprate Application (NRC ADAMS

Accession Number ML010950020), “Proposed Change Number NPF-10115-514 Increase in Reactor Power to 3438 MWt San Onofre Nuclear Generating Station Units 2 and 3”, SCE stated “ By the above reference Southern California Edison (SCE) submitted Amendment Application Numbers 207 and 192 to the facility operating licenses for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3, respectively, to increase the licensed reactor thermal power level to 3438 MWt. At 100% power operation, steam generator pressures typically vary between 800 psia and 815 psia, compared to the original nominal design operating pressure of 900 psia. Wear at tube support structures is a known degradation mechanism at SONGS. At SONGS, rapid wear was observed on tubes surrounding the stay cylinder in the center of the steam generator during the first cycle of operation. Many tubes in the most susceptible region around the stay cylinder have been preventively plugged. The first preventive plugging was done after 0.7 EFPY of operation. The preventively plugged region was expanded during the Cycle 3 outage. Typical active wear in CE designed steam generators has occurred at the support structures in the upper bundle region of the steam generator. These supports consist of diagonal straps (frequently called bat wings) and vertical strap supports. This currently active wear mechanism is influenced by both flow velocities and tube to support gap wear. The variable influenced by the proposed uprate is the inner bundle flow velocities. The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit, i.e., small perturbations of steam pressure or circulation ratio will diminish rather than grow in amplitude. The damping factors remain highly negative, at a level comparable to the current design, for all cases. Thus, the steam generators remain hydrodynamically stable for all uprate cases. Based on a projected increase of 2.3% in the secondary side fluid velocity, normal operation flow induced vibration analysis is impacted by the velocity increase. Current analysis considered that tubes with more than one consecutive inactive eggcrate were staked and plugged, and two nonconsecutive inactive eggcrates are acceptable. The Stability Ratio (SR) is defined as: SR = Veff/Vcr, where, Veff= effective velocity, Vcr = critical velocity; and Values of SR 35% of ~381 tubes in Unit 3 RSGs. Palo Verde made similar changes to their RSGs under a 50.90 License Amendment. PVNGS Generators are running after 10 years with very little tube plugging whereas the above changes in SONGS RSGs destroyed Unit 3 and crippled Unit 2 RSGs. Because of these adverse design changes, everybody is on the run: NRC Region IV, SCE, Mitsubishi, California Public Utilities Commission, Senator Barbara Boxer and Senator Dianne Feinstein. NRC Region IV, Westinghouse, AREVA, MHI, World’s Experts, SCE (Except DAB Safety Team SONGS Anonymous Insiders) are not sure whether fluid elastic instability in Unit 2 occurred or not. Southern Californians Ratepayers have lost $1 Billion in this game without electricity and now are faced with the trauma of restart of defectively-designed and degraded Unit 2 due to SCE’s continued mistakes. I am just trying to help, so please, wake up NRC Region IV and San Onofre Special Panel, Your charter is public safety and not whether SCE looses or makes money. I guarantee that SCE will make more money by admitting their mistakes and win NRC/Public Confidence by correcting their mistakes and using “Critical Questioning & Investigative Attitude” in the future. Remember, Mr. Dricks, Truth always prevails….. HAHN BABA

comment #45861 posted on 2013-01-23 10:58:20 by Moderator in response to comment #45344

Region IV used technical experts from headquarters, including the Office of Nuclear Regulatory Research, as part of the Augmented Team Inspection following the steam generator tube leak. These technical experts have continued to advise and make recommendations to the Oversight Panel, as the NRC has conducted follow-up inspections and reviews the SONGS CAL response. Before the NRC makes a restart decision, it will ensure all the appropriate discipline experts, including thermal hydraulics and materials, have reviewed their respective areas of technical expertise. Victor Dricks

comment #46078 posted on 2013-01-24 00:12:56 by in response to comment #45898

Reply to CapD: What is the DAB? Mel Silberberg

comment #45872 posted on 2013-01-23 11:35:58 by Mel Silberberg in response to comment #45861

Thank you Victor. Was the SONGS problem discussed with the ACSR? If so please send a reference to the meeting. Why wasn't their an intensive, public peer review meeting (conference) involving experts from around the world, including EPRI, comparing their analyses. The SONGS issues were so surprising - we've been using steam generators for so long, one has to suspect some new phenomena and or condition never seen before. Given the financial impact and safety significance--the public demands reassurance. Peer reviews are done for this reason. The cost of the shutdown, new generators, and replacement power cost to SCE is over a billion dollars! If I were the industry I would be concerned-- this is not a nuclear problem- but the general public doesn't know the difference. You need to answer these questions at the Public Meeting next month in Carlsbad. Mel Silberberg

comment #45984 posted on 2013-01-23 17:03:24 by CaptD

A DAB Safety Team Request to the Office of Nuclear Regulatory Research (NRR) Thermal-Hydraulic Experts. Please carefully review the SONGS Unit 2 Restart Reports (done by SCE, Westinghouse, AREVA and MHI), SCE Unit 3 Root Cause Evaluation, NRC AIT Report, ATHOS Modeling Results and Unit 2 Operational Data and then arrive: (1) At an unanimous, clear and concise conclusion whether FEI occurred in Unit 2 or not, and (2) Provide a GAP ANALYSIS (The scientific and engineering reasons why all these reports are so different) prior the February 12, 2013 NRC Public Meeting. This will be most helpful for everyone on the Special Hearing Panel and the public at large. ===> BTW: The DAB Safety Team will show you ours after the NRC shows US theirs...

comment #49562 posted on 2013-01-30 21:35:31 by CaptD

BIG TIP: The Attorney General of CA has now requested Party Status in the CPUC investigation of Edison's San Onofre Debacle!

comment #49474 posted on 2013-01-30 17:11:10 by Moderator

Note from the Moderator: Information about the NRC Inspector General's HotLine is located here: http://www.nrc.gov/insp-gen/oighotline.html . The Office of the Inspector General at NRC established the Hotline (1-800-233-3497) program to provide the NRC employee, other government employee, licensee/utility employee, contractor employee, and the public with a confidential means of reporting incidences of suspicious activity to the OIG concerning fraud, waste, abuse, and employee or management misconduct. Mismanagement of agency programs or danger to public health and safety may also be reported through the Hotline.

comment #45927 posted on 2013-01-23 14:04:25 by CaptD

From Mr. Dricks's NRC Feb. 12, 2013 Meeting notice "NRC TO MEET PUBLIC TO DISCUSS SAN ONOFRE NUCLEAR GENERATING STATION STEAM GENERATOR ISSUES" snip: " A leak in a Unit 3 steam generator tube on Jan. 31, 2012, led to the shutdown of that unit. The other reactor, Unit 2, was shut down for maintenance and refueling at the time. Subsequent inspections of the nearly new steam generators in both units found unexpected wear. Both units remain safely shut down and will not be permitted to restart until NRC has reasonable assurance they can be operated safely." To Mr. Dricks's: PLEASE define "reasonable assurance," as the Health and Safety of 8 Million people living in southern California (within 50 miles of SanO), who are depending upon the NRC to "get it right this time after failing to get it right last time," because the USA cannot afford a Trillion Dollar Eco-Disaster like Fukushima where the Japanese nuclear regulators thought they had everything covered before 3/11/11 and time proved them tragically wrong.

comment #45898 posted on 2013-01-23 13:05:25 by CaptD in response to comment #45347

Salute to Mel Silberberg for his great comment! As I have also posted, the NRC needs to populate this panel with people from outside Region IV for obvious reasons and also technical reasons as Mr. Silberberg mentions above. + Perhaps Mel would consider helping the DAB Safety Team's "Battery of Nuclear Experts", if so our contact info is listed on any of our documents posted here: https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit

comment #45902 posted on 2013-01-23 13:13:06 by CaptD in response to comment #45861

Using "technical experts from headquarters" is not the same thing as having them DIRECT this SPECIAL panel's "discovery" process! As populated now, this review panel can insure that Region IV stays in charge of its own investigation, which should not be the case, since SanO problems were caused in part by Region IV in the first place due to lax enforcement!

comment #45912 posted on 2013-01-23 13:27:11 by CaptD in response to comment #44456

To Hiddencamper Thanks for the info but I did mean that Edison "gamed" the process when they did their 50.59, for more see https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=1NpccIjm-2NhFc8uvEHT6GLPE6-LDD-aPasHtRVem3VQ

comment #48656 posted on 2013-01-29 14:21:30 by Moderator in response to comment #48318

The NRC has already conducted several reviews of the 10 CFR 50.59 documents associated with the replacement of the steam generators at SONGS. These reviews involved NRC inspectors from multiple offices including Region IV, Region II and the Office of Nuclear Reactor Regulation at NRC headquarters. The results of these reviews are contained in NRC two inspection reports that are available at http://www.nrc.gov/info-finder/reactor/songs/tube-degradation.html. [see the Augmented Inspection Team Report dated July 18, 2012, and the Augmented Inspection Team Follow-Up Report dated November 9, 2012]. It is worthy of note that the NRC staff is currently reviewing 10 CFR 50.59 documents associated with the licensee's proposed restart activities. The results of the ongoing review will be documented in a future inspection report. Victor Dricks

comment #43800 posted on 2013-01-17 22:00:44 by CaptD

SCE'S PR MACHINE IS CAPABLE OF OVERCOMING ALL HURDLES, EXCEPT GOOD SCIENCE AND SAFETY Here is much more about: NRC Violating Presidential Directive and the Public Trust https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=1QnQRbpsWgxn5xVftt-fmBBaPohF5m5OXyQgyWuz8vJI San Onofre Unit 2 Restart Decision by NRC Imminent – SCE's PR Machine Is Capable Of Overcoming ALL Hurdles, Except Good Science And Safety NRC's enforcement history, drama and pre-rehearsed tough questions, press reports, casual relationship and/or protection of SCE officials and utility biased public meetings are just old and cheap regulatory tricks that are now being used to protect the NRC's own public image and to fool the public into believing that the NRC is really concerned about public safety regarding SCE's Restart Plan. The Justice Department & NRR Officials need to set up a legal/technical taskforce to publically question Edison's design and MHI Engineer's listed below under oath regarding their: Understanding of their legal obligations under the 10 CFR 50.59 Process, Understanding of problems with the original steam generators, Critical questioning and professional/investigative skills, Efforts made in industry and academic benchmarking to identify and resolve problems with the original steam generators What part did they play in the preparation of design specifications, fabrication, computer modeling, mock-up testing, anti-vibration bar structure, and research required to prevent the adverse effects of fluid elasticity and flow-induced random vibrations in these unique San Onofre Combustion Engineering replacement generators. Any NRC decision to grant a restart of Unit 2 without a formal 50.90 licensing review along with public participation will be seen as an invitation to risk a Fukushima-type disaster happening in Southern California.

comment #53751 posted on 2013-02-08 01:11:21 by HelpAllHurtNeverBaba

Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog SPECIAL Public/NRC/SCE Awareness Series by HAHN Baba Subject: 2-6-2013: Senator Barbara Boxer Letter To The Honorable Allison M. Macfarlane (Continued) So what is new. SCE and MHI have been avoiding regulatory process since 2004 under the pretense of, “like for like.” NRC Region IV has not done anything to stop that. Now Cat is out of the bag. Senator Barbara was advised in August 2012, and a recommendation was made to form a Joint Task Force of Justice Department, Senate Committee on Environment and Public Works and NRC to investigate both SCE/MHI. But, that recommendation was swept under the rug for reasons unknown. Email To President Obama’s Campaign - info@2013pic.org Attention: Your Excellency Honorable President Barack Obama, Greatest People's President in the Modern History of United States Continued cover-up by NRC Region IV, Southern California Edison and Mitsubishi Heavy Industries in San Onofre Nuclear Generating Station Replacement Generators has become a nightmare for 8.4 Million Southern Californians. Here is a copy of a press release FYI. On Wednesday February 6, 2013 , Sen. Barbara Boxer pressed federal regulators to open an investigation at the plant after uncovering documents that she said suggest that Southern California Edison took engineering shortcuts and compromised safety. The Democratic senator said in a letter to Nuclear Regulatory Commission Chair Allison Macfarlane that a confidential report obtained by her office shows Edison and Mitsubishi Heavy Industries, the Japan-based company that built the plant’s steam generators, were aware of design problems before the equipment was installed in 2009 and 2010. Boxer, who chairs the Environment and Public Works Committee, said the report written by Mitsubishi raises concerns that Edison and its contractor rejected safety modifications and sidestepped a more rigorous safety review. “Safety, not regulatory short cuts, must be the driving factor in the design of nuclear facilities, as well as NRC’s determination on whether (San Onofre) can be restarted,” Boxer said in a letter co-signed by Rep. Edward Markey, D-Mass. Billion Dollar Safety Question is who is telling the truth? It is time now that a Joint Task Force of Justice Department, Senate Committee on Environment and Public Works and NRC ASLB & San Onofre Special Panel be created to investigate both SCE/MHI Engineers, NRC Region IV and NRC AIT Engineers involved with San Onofre and let them testify under oath. I will testify under oath to help everybody with whatever I know on the Joint Task force. 8.4 Million Southern Californians will certainly appreciate their President's help in resolving this matter of Great National Interest. FEI did not occur in Unit 2, because, (1) It was the absence of high steam dryness ALONE in Unit 2 that FEI did not occur in Unit 2 (emphasis added), and (2) Not because of the better supports (Dietrich is stating backwards and claiming incorrectly) and/or differences in fabrication, which resulted in substantially increased contact forces (reduced looseness) between tubes and AVBs for Unit 2 and prevented FEI from occurring. These supports are designed for out-of plane protection and not for-in plane protection. MHI stated, “In the design stage, MHI assumed that the tube support in the out-of-plane direction with “zero” tube-to-AVB gap in hot condition was sufficient to prevent tube from becoming fluid-elastic unstable during operation. But, the recent SONGS experience shows that the flat bar AVBs does not provide friction forces required to prevent tubes from vibrating in the in-plane direction and eventually becoming fluid-elastic unstable under high local secondary thermal-hydraulic conditions such as in the SONGS RSGs. In addition, MHI concludes that in the Unit-3 RSGs low tube and AVB fabrication dimensional dispersion causes that the tube-to-AVB contact forces are not sufficient to prevent the in-plane motion of tubes. Because of pressure from NRC and SCE and to continue Business in United States, MHI has reverted its stand from its original design position and contemporary experience. Honorable Senator Barbara Boxer is absolutely right, when she contends, “Mitsubishi and Edison were aware of safety problems with steam generators before the equipment was installed starting in 2009 but rejected some enhancements to avoid a more rigorous regulatory review. SCE and MHI accepted some adjustments to the replacement steam generators, further safety modifications were found to have “unacceptable consequences” and were rejected: “Among the difficulties associated with the potential changes was the possibility that making them could impede the ability to justify the RSG [replacement steam generator] design” without the requirement for a license amendment. The Report also indicates that SCE’s and MHI’s decision to reject additional safety modifications contributed to the faulty steam generators and the shutdown of reactor Units 2 and 3.” DAB Safety Team member warned Senator Barbara Boxer of similar concerns in August 2012, but the warning were swept under the rug for reasons unknown. DAB Safety Team findings on Unit 2 FEI and supports are consistent with the findings of AREVA, Westinghouse, John Large, SONGS RCE Anonymous Root Cause Team Member and latest research performed by Eminent Professor Michel Pettigrew and others in 2006. Therefore, SCE claims that insufficient contact forces in Unit 3 Tube-to-AVB Gaps ALONE caused tube “to” tube wear are misleading, erroneous and designed to put the blame on MHI for purposes of making SCE look good in the public’s eyes and for collecting insurance money from MHI’s manufacturing so called defects….. Thanks HAHN Baba

comment #45554 posted on 2013-01-22 11:57:38 by CaptD

The latest from DAB Safety Team on SanO: Press Release 13-01-22 ATHOS Validity Questioned, Qualifying Investigation Required https://docs.google.com/document/d/1ltCb57ciXRaOkhK1rhc2BaB0ACXf7MwcSDZZyEAkFDI/edit

comment #53587 posted on 2013-02-07 11:41:14 by HelpAllHurtNeverBaba

Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog SPECIAL Public/NRC/SCE Awareness Series by HAHN Baba Subject: 2-6-2013: Senator Barbara Boxer Letter To The Honorable Allison M. Macfarlane So what is new. SCE and MHI have been avoiding regulatory process since 2004 under the pretense of, “like for like.” NRC Region IV has not done anything to stop that. Now Cat is out of the bag. Senator Barbara was advised in August 2012, and a recommendation was made to form a Joint Task Force of Justice Department, Senate Committee on Environment and Public Works and NRC to investigate both SCE/MHI. But, that recommendation was swept under the rug for reasons unknown. Billion Dollar Safety Question is who is telling the truth? It is time now that Joint Task Force of Justice Department, Senate Committee on Environment and Public Works and NRC ASLB & San Onofre Special Panel be created to investigate both SCE/MHI Engineers, NRC Region IV and NRC AIT Engineers involved with San Onofre and let them testify under oath. I will testify under oath to help everybody with whatever I know on the Joint Task force. On Wednesday, Sen. Barbara Boxer pressed federal regulators to open an investigation at the plant after uncovering documents that she said suggest that Southern California Edison took engineering shortcuts and compromised safety. The Democratic senator said in a letter to Nuclear Regulatory Commission Chair Allison Macfarlane that a confidential report obtained by her office shows Edison and Mitsubishi Heavy Industries, the Japan-based company that built the plant's steam generators, were aware of design problems before the equipment was installed in 2009 and 2010. Boxer, who chairs the Environment and Public Works Committee, said the report written by Mitsubishi raises concerns that Edison and its contractor rejected safety modifications and sidestepped a more rigorous safety review. "Safety, not regulatory short cuts, must be the driving factor in the design of nuclear facilities, as well as NRC's determination on whether (San Onofre) can be restarted," Boxer said in a letter co-signed by Rep. Edward Markey, D-Mass. But Edison said in a statement "it is simply not accurate" to suggest the company was aware of design problems, and pointed out the equipment carried a 20-year warranty against defects. "SCE would never, and did not, install steam generators that it believed would not perform safely," the company said. Edison "sought to purchase replacement steam generators that would meet or Mitsubishi said design decisions were made "in accordance with well-established and accepted industry standards" along with a wealth of operating experience. "Nothing is more important to us than the safe design and manufacturing of nuclear-energy facilities," a company statement said. "A thorough investigation has been ongoing and will continue. We will continue cooperating fully." In a statement, the NRC said it received the letter and "will review all available information in making a judgment as to whether the plant would meet our safety standards if restart were permitted." Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog

comment #44523 posted on 2013-01-18 15:16:45 by James Greenidge

Wish nuclear opponents would just come clean and just say you're not concerned of any fix or even perfect reactors but just want them all the hell out of here as a matter of "conscience". This way meetings can deal with the more open-minded concerned who don't belittle and disparage the family-loving engineers and technicians who are investing their time and effort not only fixing reactor issues but making them even safer than they are now. Yes, safe reactors are an contradiction to hard-core antis, but the historic mortality record of nuclear reactors is unassailable and enviable, especially since it's pretty hypocritical to other energy sources slide with their score of the tens thousands killed and maimed just by accidents alone. It seems some safety concerns are barking up the wrong bogeyman to me. James Greenidge Queens NY

comment #43476 posted on 2013-01-17 17:19:31 by HELPAllHurtNeverBaba

Dear Mr. Art Howell NRC's Region IV Deputy Regional Administrator FYI and Help - Courtesy of DAB Safety Team Please read Media Alert 13-01-17 Allegation - NRC Violating President's Directive And the Public Trust and other San Onofre Papers by searching DAB Safety Team & Related Information on Google Drive. Personal Note: If you want to know more SCE Safety and Retaliation Issues and examine the evidence with my and NRC attorneys present about SCE Management, Retaliation, Safety Issues and San Onofre $1 Billion Radiation Steaming Crucible Watergate Insider Secrets, please visit me in Southern California for a Face-To-Face Technical Meeting. These concerns have already been relayed to Senator Barbara Boxer. Please feel free to send me an email helpallcqiascnp@yahoo.com. Thanks HELPAllHurtNeverBaba

comment #43980 posted on 2013-01-18 00:20:35 by HelpAllHurtNeverBaba

Chairman Allison Macfarlane said Unit 2 would not be permitted to restart unless the NRC has reasonable assurance it can be operated safely. Let us examine at the scenario below and determine whether NRC can have that reasonable assurance or not? Thanks to NRC for posting this blog. Just trying to help... HelpAllHurtNeverBaba Let us examine, what John Large, Internationally Known Scientist and Chartered Nuclear Engineer from London says about the SONGS Unit 2 Replacement Steam Generators (RSGs) AVB Structure, "It impossible to reliably predict the effectiveness of the many thousands of AVB contact points for when the tube bundle is in a hot, pressurized operational state. The combination of the omission of the in-plane AVB restraints, the unique in-plane activity levels of the SONGS RSGs, together the very demanding interpretation of the remote probe data from the cold and depressurized tube inspection, render forecasting the wear of the tubes and many thousands of restraint components when in hot and pressurized service very challenging indeed." Let us examine, what MHI says about tube-to-AVB Gaps, "The most likely cause of the observed tube-to-tube wear is multiple consecutive AVB supports becoming inactive during operation. This is attributed to redistribution of the tube-to-AVB-gaps under the fluid hydrodynamic pressure exerted on the tubes during operation. This phenomenon is called by MHI, "tube bundle flowering" and is postulated to result in a spreading of the tube U-bends in the out-of-plane direction to varying degrees based on their location in the tube bundle (the hydrodynamic pressure varies within the U bend). This tube U-bend spreading causes an increase of the tube-to-AVB gap sizes and decrease of tube-to-AVB contact forces rendering the AVB supports inactive and potentially significantly contributing to tube FEI." Let us examine, what AREVA says about tube-to-AVB Gaps, "Contact forces significantly increase at normal operating temperature and pressure due to diametric expansion of the tubes and thermal growth of the AVBs. After fluid elastic instability develops, the amplitude of in-plane motion continuously increases and the forces needed to prevent in-plane motion at any given AVB location become relatively large. Hence shortly after instability occurs, U-bends begin to swing in Mode 1 and overcome hindrance at any AVB location." Let us examine, what Westinghouse says about tube-to-AVB Gaps, "Test data shows that the onset of in-plane (IP) vibration requires much higher velocities than the onset of out-of-plane (OP) fluid-elastic excitation. Hence, a tube that may vibrate in-plane (IP) would definitely be unstable OP. A small AVB gap that would be considered active in the OP mode would also be active in the IP mode because the small gap will prevent significant in-plane motion due to lack of clearance (gap) for the combined OP and IP motions. Thus, a contact force is not required to prevent significant IP motion." Let us examine, what John Large says further, "There is no account of the changes that have been made in the evaluation of the tube structural and leakage integrity, that is from the stage of predicting those tubes at risk of TTW and other forms of wear, the tube thinning wear rates, through to the nature of the tube failure being unique to the type and extent of the wear pattern and tube thinning; and the methods of deducing, mainly by unproven inference, from the probe inspection results particularly to determine the in-plane AVB effectiveness, includes unacceptably large elements of test and experimentation that are inconsistent with the analyses and descriptions of the FSAR. I provide a number of explicit examples where I consider that the circumstances and risks accompanying the proposed restart of Unit 2 will result in unacceptable levels of test and experiment. What these World Known Experts are saying is that this degraded tube bundle cannot prevent multiple tube ruptures from fluid elastic instability as we saw by the failure of 8 tubes in Unit 3 RSGs under Main Steam Line Break (MSLB) test conditions. Main Steam Line Break (MSLB) Scenario: The most severe design basis accident to meet the SONGS Unit 2 TS 5.5.2.11.b.1 steam generator structural integrity is a MSLB at the first weld outside containment. This assumption minimizes the flow resistance between the break and the affected SG and maximizes the mass & energy (M&E) release. The analyses focus on M&E releases at licensed Rated Thermal Power (RTP or 100% Power). The outside containment case includes the assumption that the main steam isolation valve (MSIV) in the steam line with the least flow resistance fails to close following the main steam isolation signal (MSIS). This assumption maximizes the M&E release during a MSLB outside of the containment. Super-heating within the SG initiates upon U-tube uncovery as specified in the NRC Information Notice 84-90. The turbine stop valves are assumed to close instantaneously at the time of the reactor trip. This assumption is conservative for a MSLB event because the entire steam inventory at the time of reactor trip is assumed to be forced out the break in 300 seconds or 5 minutes. No Operator action outside Control Room can be credited, if it takes less than 30 minutes. The depressurization of the non-isolable steam generator would result in 100% void fractions in the degraded Unit 2 U-Tube bundle due to instant flashing of the sub-cooled 440 degrees Fahrenheit feedwater into steam. This condition of ZERO Water in the steam generators would cause fluid elastic instability (FEI), flow-induced random vibrations and excessive hydrodynamic pressures (Mitsubishi Flowering Effect). The force of the flashing steam would create high-energy jets, lifting loose parts and debris present in the steam generator, which would do additional damage by cutting holes into the already degraded tubes and creating additional loading (See Note A below) on the tube support plates (TSPs) due to heavy build-up of deposits on trefoil/quadrifoil-shaped holes from SG blowdown and crack the high cycle fatigued U-bend tubes not supported by Anti-Vibration Bars (AVB). These cumulative adverse conditions in all likelihood would result in a massive cascading of RSG's tube failures (tubes would excessively rattle or vibrate, hitting other tubes with violent impacts) due to extremely low tube-to-tube clearances and no effective or non-existent in-plane anti-vibration bar support protection system. This jackhammering effect would involve hundreds of degraded active SG tubes along with all the inactive (plugged /unstabilized) tubes causing a catastrophic amount of simultaneous tube leaks/ruptures. Under this adverse scenario, approximately 60 tons of very hot high-pressure radioactive reactor coolant would leak into the secondary system. The release of this amount of radioactive primary coolant, along with an additional approximately 200 tons of steam in the first five to fifteen minutes from a broken steam line would EXCEED the SONGS NRC approved offsite radiological release doses safety margins based on assumption of a single tube rupture in the SONGS FSAR. So, in essence, these RSG's are like loaded guns, or a Fukushima-type nuclear accident, waiting to happen. Any failure under these conditions would allow significant amounts of radiation to escape to the atmosphere and a major Loss of Coolant Accident (LOCA) could easily result causing much wider radiological consequences and even a potential nuclear meltdown of the reactor. SCE states, "A MSLB alone does not generate sufficient differential pressure to cause tube rupture. The differential pressure across the SG tubes necessary to cause a rupture will not occur if operators (See Note B below) prevent RCS re-pressurization in accordance with Emergency Operating Instructions." SCE's suggested DID Actions and proven unreliable operator actions to detect a leak and/or to re-pressurize the steam generators as claimed by Edison are not practical to stop a major nuclear accident from occurring in Unit 2 in the first 5-15 minutes of a MSLB during the proposed 5-month trial period. NOTES: A. Plugging of the at-risk tubes is not a satisfactory solution because it is the retainer bar that vibrates via random fluid flow processes at sub FEI critical velocity levels - these are likely to continue in play or, indeed, exacerbate at the proposed U2 restart at 70% power, leading to through-tube abrasion, the detachment of tube fragments, lodging at other unplugged and in-service tube localities, resulting in the so-called 'foreign object' tube wear. This additional loading would exceed: (1) the safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials, and (2) significantly affect burst or collapse pressures determined and assessed in combination with the loads due to a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. [emphasis added B. SCE's suggested "defense-in-depth" actions are insufficient to stop multiple tube ruptures due to the short duration of a main steam line beak event. Human performance weaknesses, such as mis-diagnoses, substantial delays in isolating the faulted steam generator, communication errors and delayed initiation of the residual heat removal system, have been identified in past events at SONGS and other US Nuclear Power Plants. The events also involved unnecessary radiation releases, lack of RCS subcooled margin, excessive RCS cooldown rates, and overfilling the SG because of human or procedural problems. TRILLION DOLLAR ECONOMIC-ENVIRONMENT-HUMAN DISASTER QUIZ: If NRC Region IV Special Technical Panel can determine whether Southern Californians have reasonable assurance that a Fukushima will not occur by granting Unit 2 Restart Permission to Profit-Motivated Gambler SCE? If Fukushima occurs, who is liable, Nuclear Utilities, Insurance Carriers, Federal Government, State of California, CPUC, NRC Commissioners, NRC Region IV, EIX/SCE Shareholders & Employees or Affected Southern Californians?

comment #45577 posted on 2013-01-22 12:27:49 by CaptD

SanO Nuclear Denial*? Perhaps this panel will also explain why "severe accident" is not even listed in it's 130 page NRC Collection of Abbreviations, especially since there are two classes of accident: postulated accidents and severe accidents. * http://is.gd/XPjMd0 The illogical belief that Nature cannot destroy any land based nuclear reactor, any place anytime 24/7/365!

comment #53342 posted on 2013-02-06 22:40:56 by Help Ever Hurt Never Baba

The Research by the World's Number 1 Expert in 2006 shows that flat bars are not effective to protect the SG tubes from the adverse effects of Fluid Elastic instability and Low Frequency reatainer bars can damage tubes from turbulence induced random vibrations. According to SONGS insiders, SONGS RCE Team had access to this information. But the results were both ignored by SCE and MHi. Transparenecy and accountabilty is the rule of law for a licensee and its contractor for nuclear safety. if this information true, it is a blatant violation of federal regulations and public trust by SCE and MHI. Senator Barbara Boxer was warned of these types of cover ups in August 2012 and investigation by Justice Department was requested. These recommendations were swept under the rug for reasons unknown. Thanks to Mr. Victor Dricks for posting this blog.

comment #43836 posted on 2013-01-17 22:24:30 by CaptD

Lets be scientifically rigorous, SCE's Richard St. Onge has as much of a chance of explaining how FEI destroyed Unit 2 and 3 at SanO by pointing to their poorly in-hous designed 620 ton replacement steam generator, as NRC Chairman Allison Macfarlane would have of explaining how dip/slip effects earthquakes by pointing to the earth! This photo is an insult to the intelligence of your readers and the Director of the NRC, who happens to be a World Class Geologist...

comment #48318 posted on 2013-01-29 01:04:55 by HelpAllHurtNeverBaba

To: NRC Moderator Mr. Victor Dricks, Senior Public Affairs Officer, NRC Region IV Request for independent re-review of SONGS 50.59 Screen/Evaluation by NRC Region II - Please send me an email after you complete the review ASAP. These guys who performed the screen and evaluations are very close friends of mine and I want to make sure they were on the right track. Trying to help my friends and NRC Region IV. Thanks... HAHN Baba

comment #44454 posted on 2013-01-18 10:54:11 by CaptD in response to comment #43393

It's thinking like that that has gotten Edison and too many of the NRC's regulators into hot water! BTW in answer to your questions: 1. One RSG failed in less than a year of operation (starting from NEW) and now has more damaged, worn and plugged tubes that the entire rest of the US Nuclear fleet. 2. Both Unit 2 and Unit 3 have a unknown amount of wear because Edison has not visually inspected all the thousands of tubes using the best technology to insure safety, because then they would be forced to exceed their tube plugging limitation! 3. This is nothing but using San as a TEST site instead of a proven safe reactor/RSG as the law demands. 4. Even thinking 50-50, either it will or will not have a nuclear incident, it's N☢T worth the Gamble! 5. Wrong, worse case is a Fukushima-type event/disaster after a MSLB... That is factual! 6. We calculated that within 5 minutes, a cascading tube failure would doom the reactor core, because of loos of coolant which would be vented to the atmosphere, all beyond the ability of the Operators to prevent!

comment #44456 posted on 2013-01-18 10:56:31 by CaptD in response to comment #43549

The more that take interest and demand an open 50.59 process before any restart decision, the safer all of California and the rest of the US will be...

comment #43393 posted on 2013-01-17 15:54:58 by joffan7

OK, let's be efficient here. I can do this panel's remaining work in one comment. 1. Past performance of this steam generator: no leaks 2. Past work: block tubes showing any sign of wear 3. Proposal: operate the reactor for a shorter period than produced the previous wear, reevaluate afterwards 4. Likely outcome: safe operation with intact steam generator due to points 1,2,3 @ probability > 99% 5. Worst case outcome: tube leak/break in steam generator 6. Worst case consequence: immediate detection, reactor shutdown, no safety implications Conclusion: Restart SONGS 2 immediately.

comment #48791 posted on 2013-01-29 18:54:36 by CaptD

Allegation – NRC Region IV Violating Presidential Directive and the Public Trust ttps://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=1S3tUKp-sV-rS2OFMY0Z-A1H4BJ4Ha1ftiow7577tsdY snip: SONGS UNIT 3 RSG ROOT CAUSE: It appears that Complacent SCE and Inexperienced MHI Engineers did not perform proper academic research and industry benchmarking about the potential adverse consequences of the reduction of original CE steam generator pressures from 900 psi to say, 800 psi on fluid elastic instability and flow-induced vibrations. These lower secondary steam operating pressures (800-833 psia) are the primary cause for shortening the life of SONGS Original Combustion Engineering Generators due to increased tube wear and plugging caused by flow-induced random vibrations and destruction of SONGS Unit 3 Replacement Steam Generators due to flow-induced random vibrations, Mitsubishi flowering effects and steam voids or steam dry-outs (AKA fluid elastic instability). In addition, SCE Engineers prepared a defective 10 CFR 50.59 Evaluation and design specifications, which were not challenged by MHI, and/or adequately reviewed by NRC Region IV. MHI at the direction of SCE Engineers made numerous untested and unanalyzed design changes to the steam generators under the pretense of “like for like”, and even the NRC’s Region IV administrator Elmo Collins said, “The guts of the machinery look …. Different.”

comment #49810 posted on 2013-01-31 08:03:14 by CaptD in response to comment #48656

Salute to Mel Silberberg, If you do reach the Exec. Director of ACRS, please tell him to contact the DAB Safety Team, we have posted more factual data/information* about San Onofre's FEI problems than anybody else! The DAB Safety Team's documents explain in detail why a SONGS restart is unsafe at any power level, especially without a Full/Thorough/Transparent NRC 50.90 License Amendment and Evidentiary Public Hearings. For much more from the DAB Safety Team, please visit the link* below. * https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?pli=1

comment #50315 posted on 2013-01-31 22:49:34 by Mel Silberberg

I am trying to reach Dr. Hackett. Mel Silberberg, USNRC Retired

comment #50490 posted on 2013-02-01 02:30:33 by HelpAllHurtNeverBaba

San Onofre Special Public/NRC/SCE Awareness Series by HAHN Baba Critical Questioning & Investigative Attitude Quiz An unnamed US Nuclear Power Plant increased its power by 7.5 % and increased the heat transfer surface area of its Replacement steam Generators by 25%. The RSG design addressed Fluid Elastic Instability, Flow -induced Random Vibrations and compared the results with Several Operating Reactors. It will be interesting to find out if NRC region IV can answer what was the name of that US Plant and how the RSG heat transfer surface area was increased. Did this plant apply for a 50.90 License Amendment and which NRC Branch checked the results of Thermal-Hydraulic Modeling/FEI/FIRV Vibrations? Thanks to Mr. Victor Dricks for Posting this Blog.

comment #43323 posted on 2013-01-17 11:42:53 by CaptD

The above Panel should N☢T be co-chaired by anyone in Region IV, since their supervision of Edison has been called into question and the panel should include at least one and preferably two outside experts to insure that this HISTORIC NRC/NRR Panel is not just covering up for the NRC (and Edison) to protect its own public image! FEI does not care about NRC internal politics, nor does it follow inter-office memo's or yield to graft.

comment #53429 posted on 2013-02-07 03:30:31 by HelpAllHurtNeverBaba

Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog SPECIAL Public/NRC/SCE Awareness Series by HAHN Baba Subject: 2-6-2013: Senator Barbara Boxer Letter To The Honorable Allison M. Macfarlane We have become aware of new information contained in a 2012 Mitsubishi Heavy Industries (MHI) document entitled “Root Cause Analysis Report for tube wear identified in the Unit 2 and Unit 3 Steam Generators of San Onofre Generating Station” (Report). The Report indicates that Southern California Edison (SCE) and MHI were aware of serious problems with the design of San Onofre nuclear power plant’s replacement steam generators before they were installed. Further, the Report asserts that SCE and MHI rejected enhanced safety modifications and avoided triggering a more rigorous license amendment and safety review process. For example, the Report states that although SCE and MHI accepted some adjustments to the replacement steam generators, further safety modifications were found to have “unacceptable consequences” and were rejected: “Among the difficulties associated with the potential changes was the possibility that making them could impede the ability to justify the RSG [replacement steam generator] design” without the requirement for a license amendment. The Report also indicates that SCE’s and MHI’s decision to reject additional safety modifications contributed to the faulty steam generators and the shutdown of reactor Units 2 and 3. This newly-obtained information concerns us greatly, and we urge the NRC to immediately conduct a thorough investigation into whether SCE and MHI did in fact fail to make needed safety enhancements to avoid the license amendment process. All people in our nation, including the 8.7 million people who live within 50 miles of the San Onofre plant, must have confidence in the NRC’s commitment to put safety before any other concern. We believe this alarming Report raises serious concerns about SCE’s and MHI’s past actions. Safety, not regulatory short cuts, must be the driving factor in the design of nuclear facilities, as well as NRC’s determination on whether Units 2 and 3 can be restarted. So what is new. SCE and MHI have been avoiding regulatory process since 2004 under the pretense of, “like for like.” NRC Region IV has not done anything to stop that. Now Cat is out of the bag. Senator Barbara was advised in August 2012, and a recommendation was made to form a Joint Task Force of Justice Department, Senate Committee on Environment and Public Works and NRC to investigate both SCE/MHI. But, that recommendation was swept under the rug for reasons unknown. Anyhow, the bottom line is neither, SCE nor MHI have the knowhow to design/build a CE Replacement Steam Generator. That being said, only Westinghouse has the skills and technology to design and build a CE Replacement Steam Generator. SCE was told that in June of 2012. But the problem is nobody listens, until it is too late. For example, Westinghouse/Combustion Engineering designed several CE Replacement Generators in 2000-2005 (e.g., PVNGS 1, 2, 3, ANO-2, etc.), which are running successfully. NRC Region IV licensed these generators with assistance from NRC Commission under a 50.90 process with a “Critical Questioning and Investigative Attitude.” During San Onofre Replacement Generator Design, Installation & Accident Investigation Process, NRC Region IV has been acting as a silent observer going along with SCE instead of a strict regulator for reasons unknown. AREVA says in its Operational Assessment, ” Weaver and Schneider [16], in 1983, examined the flow induced response of heat exchanger U-tubes with flat bar supports. It is worth quoting the first conclusion of their paper: “The effect of flat bar supports with small clearance is to act as apparent nodal points for flow-induced tube response. They not only prevented the out-of-plane mode as expected but also the in-plane modes. No in-plane instabilities were observed, even when the flow velocity was increased to three times that expected to cause instability in the apparently unsupported first in-plane mode.” Additionally, in an effort to encourage the development of in-plane instability, Weaver and Schneider substantially increased the clearances between flat bar supports and U-tubes, but no in-plane instability was observed. Other investigators, notably Westinghouse, have deliberately searched for in-plane instability with only support from flat bars and have not detected the phenomena. However in 2005, Janzen, Hagburg, Pettigrew and Taylor reported in-plane instability. The abstract to their paper states, “For the first time in a U-bend tube bundle with liquid or two-phase flow, instability was observed in both the out-of-plane and in-plane direction.” A document published in 2006, “Fluid-elastic instability of an array of tubes preferentially flexible in the flow direction subjected to two-phase cross flow. (http://yakari.polytechnique.fr/people/revio/masters_research_subject.html) by Violette R., Pettigrew M. J. & Mureithi N. W. stated, “In nuclear power plant steam generators, U-tubes are very susceptible to undergo fluid elastic instability because of the high velocity of the two-phase mixture flow in the U-tube region and also because of their low natural frequencies in their out of plane modes. In nuclear power plant steam generator design, flat bar supports have been introduced in order to restrain vibrations of the U-tubes in the out of plane direction. Since those supports are not as effective in restraining the in-plane vibrations of the tubes, there is a clear need to verify if fluid elastic instability can occur for a cluster of cylinders preferentially flexible in the flow direction.” Retainer bars also suffer with similar problems. SONGS Root cause team members told SCE Management about the problems with ineffective flat bars, but no body listed and they kept repeating only one phrase, "insufficient tube-AVB contact forces caused FEI. Westinghouse and AREVA said, Guys, Once FEI starts, contact forces do not count. The point is when you are designing and building such a complex steam generator as a San Onofre replacement steam generator, whether it is a designer or manufacturer, one is supposed to keep up with the latest university research and industry benchmarking. It is absolutely clear that SCE and MHI did not do that. They both broke the federal regulations, public trust, wasted 1 Billion Dollars of Rate Payer’s Money and almost created twin accidents. What is the use of crying now. It appears, now NRC is going on behalf of SCE to Japan to supervise MHI’s quality assurance activities associated with the mock-up and testing of re-designed anti-vibration bars that may be used as a long-term repair of both Unit 2 and Unit 3 San Onofre Nuclear Generating Station (SONGS) steam generators. As I have said with John Large, Arnie Gundersen, Professor Daniel Hirsch, David Lochbaum, and dozen other anonymous steam generator experts and DAB Safety Team before, these defectively-designed and degraded generators will not withstand a MSLB or other anticipated transients due to FEI even at 70% power. The bottom line is that MHI was a subcontractor to Westinghouse and they make huge claims, but really do not how to design a CE Replacement Generator. SCE also makes huge claims and preaches safety sermons, but is not a steam generator designer. So NRC is wasting their valuable time with SCE and MHI and wasting money of Southern California’s Rate Payers and putting their safety on line. So I request NRC Chairman, NRC ASLB & San Onofre Special Panel Members, Please tell Westinghouse to build replacement steam generators for San Onofre and tell Ted Craver and Pete Dietrich to fire the SCE Retaliating and inefficient management, work on producing safe electricity and build public trust and respect both for SCE and NRC. Thanks once again to Mr. Victor Dricks for posting this blog.

comment #44532 posted on 2013-01-18 15:55:03 by joffan7 in response to comment #44454

It's sloppy reading like that that makes your opinions so mistaken. I didn't ask any questions, so you can't possibly answer them. All your responses were in error. 1. Unit 2 steam generator has no leaks, exactly as I said. 2. All tubes have been inspected and tubes with wear blocked. Your nonsensical "visual inspection" is not an option and in any case is not the best technology for assessment. 3. Undertaking a test run, especially one with no adverse consequences, is the rational way to proceed to improve understanding of the situation and possible future scenarios. 4. Understanding likelihood is an essential part of risk management. 5. A leak or break in steam generator tubes has no prospect of causing a loss of coolant accident, no matter what fantasy incident you decide to give an acronym to. Adding "that is factual" to your daydreams does not make it any more likely. 6. Your "calculations" are immaterial; reality dictates that no such outcome will occur.

comment #43549 posted on 2013-01-17 19:42:37 by richard123456columbia

S0 NRC does not want to shut it down, they hand it off to others that will and wipe their hands of it, cowards. We now know they will allow it, what a scam.

comment #43545 posted on 2013-01-17 19:31:22 by Ace Hoffman

Gee, it looks so shiny and new you can see their reflections! And why does St. Onge have different color booties on? Is that like a leadership color or something? I too believe an unbiased review team shouldn't have any Region IV NRC personnel on it. They've already shown gross incompetence, industry bias, and unsound judgement. We don't need any more of that here. The AIT already utterly failed to find the root cause, and failed to declare Units 2 and 3 inoperable as they should have, thus allowing SCE to proceed to ask for a restart license -- maybe even the same lousy engineers and executives who built/approved the first failed redesign. It makes further sense to have non-region IV personnel for the simple reason that this problem -- FEI -- potentially affects the entire PWR nuclear industry, so they might as well get used to it. Other reactors (and their inspectors) should also be reanalyzing what might happen in a MSLB condition with degraded tubes, or in any event -- design basis or other -- that ruptures multiple SG tubes. Boeing ain't got half the problems our nuclear fleet has -- and by the way, what kind of batteries do we use at SanO???

comment #46572 posted on 2013-01-25 05:22:58 by Dee

As a professional for many years in manufacturing quality assurance, the first thing that comes to mind is effective root cause analysis. Have all the factors relating to the root cause of the problem been solidly determined? And if so, has this potential for failure been examined at all other plants that might have similar equipment setups? Has a failure mode and effects anaysis (FMEA) been conducted to ensure that all potential aspects of failure are considered for retrofitting, including the potential that something at the plant contributed to the failures? As another poster opined, over 8 million people live in the area. I think root cause analysis and FMEA study are crucial pieces to help ensure the safety of the plant and the surrounding population.

comment #51543 posted on 2013-02-03 02:21:23 by HelpAllHurtNeverBaba

San Onofre Special Public/NRC/SCE Awareness Series by HAHN Baba Promoting Critical Questioning & Investigative Attitude Edison has said exhaustive research by its team of global experts demonstrates the safety of what it calls a conservative plan to re-open SONGS Unit 2. NRC AIT Team/Edison and its team of global experts are not sure among themselves, whether fluid elastic instability occurred in Unit 2. San Onofre NRC AIT Report, SCE Unit 3 Cause Evaluation, SCE Unit 2 Return to Service Reports, SCE Response to NRR RAIs, San Onofre Special Tube Inspection Reports and 10 CFR 50.59/FSAR Justifications need to be thoroughly reviewed and a GAP Analysis prepared by brilliant NRC NRR, Civil, Mechanical, Chemical, Materials, Structural, Electrical/I&C, T/H Engineers, Computer Modeling and San Onofre Special NRC Panel Members. San Onofre Special NRC Panel Members need to make accurate and precise engineering decisions based on validated and auditable facts in accordance with Honorable and Respected Dr. McFarlane's High Standards. These decisions have to be made without any political/financial/time pressures from EIX/SCE Officers, CPUC Chairman, NRC Commissioners, Pro-SCE Politicians, Attorneys or Industry Lobbyists. Ex NRC Branch Chiefs (Dr. Joram Hopenfeld, Mel Silberberg , etc.), Anonymous "Critical Questioning & Investigative Attitude" Genius NRC Branch Chief, US Public, San Onofre Workers and Southern Californians would appreciate San Onofre Special NRC Panel Members "Critical Questioning & Investigative Attitude", True, Unbiased and Diligent Public Safety Efforts. More and more Southern Californians, Cities, Businesses, School Districts are joining every day the chorus to press Federal Regulators to hold a trial-like hearing before deciding whether the San Onofre nuclear plant is safe to reopen. Newly elected San Diego Congressman Vargas said, "When he was an assemblyman he questioned industry executives under oath during the California's energy crisis. Vargas said that process could be effective for San Onofre as long as those questioning majority owner Southern California Edison executives know what they're talking about. "Get experts in there," Vargas said. "To ask them true questions: is it really safe do you really have this under control if not why are you firing it up? Makes no sense. The only reason they're doing this is they want to get some money and if it sits vacant for a long time they actually can't recoup their investment." Federal Regulators have no choice to abide by the wishes of Southern Californians, because they pay for the cost of San Onofre and their safety is at risk. SCE, its global experts and NRC have nothing at risk in this unapproved and potentially lethal experiment. Thanks to Mr. Victor Dricks for Posting this Blog.

comment #45540 posted on 2013-01-22 11:30:53 by CaptD

Question: If Dan Dorman, is the deputy director for engineering and CORPORATE support in the Office of Nuclear Reactor Regulation (NRR), who at the NRR is tasked with providing PUBLIC SUPPORT?

comment #45697 posted on 2013-01-22 22:55:25 by HelpAllHurtNeverBaba in response to comment #45323

Respected Mr. Mel Silberberg, NRC RES Retired; former Chief of Severe Accident Research Branch ... I totally agree with your comments. Just as a reminder.... NRC website states, "As an independent regulatory agency that prides itself on openness, the U.S. Nuclear Regulatory Commission (NRC) is pleased to take an active role in President Barack Obama's Open Government Initiative, with its focus on open, accountable, and accessible government. The NRC has a long history of, and commitment to, transparency, participation, and collaboration in our regulatory activities." During FOE presentation: Top NRC official fell asleep during presentation — "His eyes were rolling back and his head was bobbling like a little bobble doll" — Process designed to freeze public out. This type of behavior during public presentations on matters of life and death for Southern Californians conflicts with NRC's commitment to participation in regulatory activities and President Barack Obama's Open Government Initiative, with its focus on open, accountable, and accessible government. This is one example, so what is new. NRC needs to shift gears and take a very aggressive, prudent and super-conservative approach, commensurate with its authority granted by the Public, President Obama and United States Congress, when it comes to SCE and San Onofre's profit-motivated and repeatedly dangerous public safety-ventures.

comment #48280 posted on 2013-01-28 22:05:33 by HelpAllHurtNeverBaba

Special Thanks to NRC Moderator Mr. Victor Dricks, Senior Public Affairs Officer, NRC Region IV for Posting this Blog Special Public Awareness Series – US #1 Nuclear Safety Concern Contrary to what the PUC news release led the public to believe the PUC issued a "scoping" memorandum today limiting the review of San Onofre issues to those helpful to SCE and hurtful The scoping memo makes a mockery of the PUC "investigation" because it allows only a very limited review of the issues: (1) assessing the reasonableness of SCE's actions and expenditures after the outage; (2) whether SCE's 2012 expenditures for SONGs was reasonable; (3) the reasonableness of SCE's expenditures for community outreach; and (4) whether SCE should refund any money they were allowed to keep under the General Rate Case issued in December 2012. Here is what will not be allowed: (1) whether SCE was imprudent and unreasonable in spending $800 million for the 4 new generators to replace the previous generators which tube problems, when the new generators had tube problems worse than those replaced; (2) whether the 4 generators should be taken out of the rate base. The Scoping Order does not address the first question and pushes off the second to some undetermined time in the future. The PUC has mislead the People of California by issuing a news release announcing an investigation while issuing an order that does not permit a reasonable investigation. It is clear that the PUC has decided to get San Onofre back in operation as soon as possible. The PUC "investigation" is nothing more than a cynical public relations stunt.

comment #48250 posted on 2013-01-28 20:19:06 by HelpAllHurtNeverBaba

Special Thanks to NRC Moderator Mr. Victor Dricks, Senior Public Affairs Officer, NRC Region IV for Posting this Blog - Special Public Awareness Series – US #1 Nuclear Safety Concern Portions of the following information have been extracted from the DAB Safety Team Reports (Search Google Drive for DAB Safety Team & Related Info). It is the DAB Safety Team's goal to help educate both the NRC and the Public by providing unbiased, logical and factual information in order to help assess the real dangers of any San Onofre Unit 2 restart. According to Press Reports and San Onofre Insiders, Unit 2 permission for restart by the NRC is imminent yet the REAL Root Cause for the $1 Billion destruction of Units 2 and 3 RSGs (Including equipment cost and expenses) has not yet even been determined. The Public does not know the status of SCE's ongoing cause evaluations, SCE's response to 32 NRR's RAI's and NRC's Special San Onofre Inspections. We like to remind NRC San Onofre Special Panel, what NRC Chairman Macfarlane said during her recent Fukushima Trip, "Regulators may need to be 'buffered' from political winds, but they need to be fully subjected to the pressure of scientific and engineering truth and cannot be allowed to make decisions or order actions that are 'independent' of facts." The NRC rush to a faulty judgment cannot be allowed to compromise Public Safety just to please SCE, as this conflicts with President Obama's Policy, the new NRC Chairman's Standards and the advice of NRC retired Branch Chiefs who have spoken out. Comments – SONGS Unit 2 Restart Reports Contradicting, Confusing, Inconclusive, Smoking Mirrors, Inconsistent and Unacceptable PROBBABLE ROOT CAUSE: Lack of "Critical Questioning & Investigative Attitude" of SCE Supplied Operational Data by Westinghouse, AREVA, MHI and Other World's Leading Experts - Public to Judge for themselves .C. Let us now examine the other differences between Unit 2 and Unit 3's Operational Factors, which were significant contributors to the "fluid-elastic instability" in San Onofre Unit 3 and the tube-to-tube wear resulting in the tube leak. C.1 – Adverse Design/Operational Factors responsible for Fluid Elastic Instability: Low steam generator pressures (SONGS RSGs range 800-850 psi, the primary cause of the onset of severe vibrations) allow the onset of FEI, whereby U-tube bundle tubes start vibrating with very large amplitudes in the in-plane directions. Extremely hot and vibrating tubes need a little amount of water (aka damping, 1.5% water, steam-water mixture vapor Fraction 99.5%). Without the water, the extremely hot and vibrating tubes cannot dissipate their energy. In effect, one unstable tube drives its neighbor to instability through repeated violent impact events which causes tube leakage, tube failures at MSLB test conditions and/or unprecedented tube-tube wear, Tube-to-AVB/Tube Support Plates wear, as we saw in San Onofre Unit 3. So in review, due to narrow tube pitch to tube diameter, tube natural frequency, low tube clearances, in certain portions of the RSGs U-tubes bundle, fluid velocities exceed the critical velocities due to extremely high steam flows (100% power conditions). These high fluid velocities cause U-tubes to vibrate with very large amplitudes in the in-plane direction and literally hit other tubes with repeated and violent impacts. Due to lower secondary steam operating pressures (required to generate more heat, electricity and profits) and excessive pressure drops due to high flows and velocities, steam saturation temperature drops. This lowering of steam saturation temperature combined with high heat flux in the hot leg side of the U-tube bundle causes steam dry-outs to form (Vapor fraction >99%), known as "NO Effective Thin Tube Film Damping." Thin film damping refers to the tendency of the steam inside the generators to create a thin film of water between the RSG tubes and the support structures and each other. That film is enough to help keep the tubes from vibrating with large amplitudes, hitting other tubes violently, and to protect the Anti-Vibration Bar support structures and maintain the tube-to-AVB gaps and contact forces. These adverse conditions in Unit 2 at 70% power operation (RTP) with the present defective design and degraded RSGs, known as fluid elastic instability (Tube-to-Tube Wear, or TTW) can lead to rapid U-tube failure from fatigue or tube-to-tube wear in Unit 2 due to a main steam line break as seen in Unit 3's RSG's. In summary, FEI is a phenomenon where due to San Onofre RSGs design intended for high steam flows causes the tubes to vibrate with increasingly larger amplitudes due to the fluid effective flow velocity exceeding its specific limit (critical velocity) for a given tube and its supporting conditions and a given thermal hydraulic environment. This occurs when the amount of energy imparted on the tube by the fluid is greater than the amount of energy that the tube can dissipate back to the fluid and to the supports. The lack of Nucleate boiling on the tube surface or absence of water is found to have a destabilizing effect on fluid-elastic stability. C.2 – Unit 2 FEI Conflicting Operational Data • NRC AIT Report SG Secondary U2/3 Pressure Range 833 – 942 psi • SCE RCE SG Secondary U2/3 Pressure – 833 psi • RCE Team Anonymous Member – Unit 2 SG Secondary Pressure 863 psi • SONGS SG System Description Unit 2 SG Pressure Range 892 – 942 psi • Westinghouse OA SG Secondary U2/3 Pressure ~ 838 psi, Void Fraction 99.55% • SCE Enclosure 2, MHI ATHOS results – U2/3 Void Fraction 99.6% • SCE Enclosure 2, Independent Expert results – ATHOS U2/3 Void Fraction 99.4% • DAB Safety Team SG Secondary U2 Pressure 863 -942 psi, Void Fraction 96-98% • SONGS Plant Daily Briefing Unit 3 Electrical Generation – 1186 MWe • SONGS Plant Daily Briefing Unit 2 Electrical Generation – 1183 MWe C.3 – Unit 2 FEI Conclusions C.3.1 – NRC AIT Report – Operational Differences between U2/3 – The NRC analysis indicated a correlation with the tube-to-tube wear based on a combination of high void fraction and high steam velocities. It should be noted that the traditional forcing function, fluid velocity squared times density, does not show good agreement with the tube-to-tube wear patterns. This indicated that the high quality steam fluid velocities and high void fraction may be sufficiently high to cause conditions in the generators conducive for onset of fluid-elastic instability. The ATHOS code predicted regions of high void fraction and high steam velocities are super-imposed with tube-to-tube wear indications from Unit 3 steam generator 3E0-88 The above analyses apply equally to Units 2 and 3, so it does not explain why the accelerated fluid-elastic instability wear damage was significantly greater in Unit 3steam generators. The result of the independent NRC thermal-hydraulic analysis indicated that differences in the actual operation between units and/or individual steam generators had an insignificant impact on the results and in fact, the team did not identify any changes in steam velocities or void fractions that could attribute to the differences in tube wear between the units or steam generators. C.3.2 – SCE Unit 2 Restart Report Enclosure 2 Conclusions – Because of the similarities in design between the Unit 2 and 3 RSGs, it was concluded that FEI in the in-plane direction was also the cause of the TTW in Unit 2. C.3.3 – SCE U2 FEI SONGS RCE Team Anonymous Member Conclusions – FEI did not occur in Unit 2. C.3.4 – Westinghouse OA Conclusions: (a) An evaluation of the tube-to-tube wear reported in two tubes in SG 2E089 showed that, most likely, the wear did not result from in-plane vibration of the tubes since all available eddy current data clearly support the analytical results that in-plane vibration could not have occurred in these tubes, and (b) Operational data – Westinghouse ATHOS Model shows no operational differences in Units 2 & 3 (void fraction ~99.6%) and then Westinghouse says in (a) above that FEI did not occur in Unit 2. Westinghouse is contradicting its own statement. C.3.5 – AREVA OA Conclusions – Based on the extremely comprehensive evaluation of both Units, supplemented by thermal hydraulic and FIV analysis, assuming, a priori, that TTW via in-plane fluid-elastic instability cannot develop in Unit 2 would be inappropriate. C.3.6 – John Large States, “I note here that there are three clear conflicts of findings between the OAs: From AREVA that AVB-to-tube and TTW result from in-plane FEI, contrasted to Westinghouse that there is no in-plane FEI but most probably it was out-of-plane FEI, and from MHI that certain AVB-to-tube wear results in the absence of in-plane FEI from just turbulent flow. My opinion is that such conflicting disagreement over the cause of TTW reflects poorly on the depth of understanding of the crucially important FEI issue by each of these SCE consultants and the designer/manufacturer of the RSGs.” C.3.7 – DAB Safety Team Conclusions – Due to higher SG pressure (Range 863 – 942 psi) and lower thermal megawatts as compared to Unit 3, FEI did not occur in Unit 2. This is consistent with the position of RCE Team Anonymous Member. The NRC AIT Report, SCE, Westinghouse, MHI, Independent Expert and AREVA conclusions on Unit 2 FEI are Contradicting, Confusing, Inconclusive, Full of Smoking Mirrors, Inconsistent and Unacceptable C.3.8 – The NRC San Onofre Special Review Panel should direct other branches within the NRC (NRC-RES and/or the ACRS) to review the above data without any prior “turf” bias and present their findings to the public for review and comment prior to any restart decision being made by the NRC.

comment #53252 posted on 2013-02-06 17:11:56 by CaptD

Big News ==> New Letter to NRC from Boxer ==> Investigate SCE and MHI http://campaign.r20.constantcontact.com/render?llr=4xdmoojab&v=001C8QXJYUMj5dcQ4dDSveBDByVNMmLB65YxIzi0_9_gVamJ_ZETvM2v1U38RPd6IPnB2rd_cOhapemnwrqO-lgo9IEGBbdZbRfMYrXw-6g7aZn-hO-WNEs7A%3D%3D …

comment #48127 posted on 2013-01-28 13:31:57 by HelpAllHurtNeverBaba in response to comment #45344

Hi Mr. Steinberg, Please See DAB Safety Team Media Alert 13-01-28 Allegations 1. NRC AIT Report Incomplete, Inconclusive, Inconsistent and Unacceptable 2. SONGS UNIT 3 RSG REAL ROOT CAUSE: Lack of “Critical Questioning & Investigative Attitude” by SCE, MHI and NRC Region IV and AIT Team. Google Drive - DAB Safety Team & Related Info Share ... docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4...

comment #52327 posted on 2013-02-04 21:26:09 by HelpAllHurtNeverBaba

Special Thanks to NRC and Moderator Mr. Victor Dricks for posting this blog SPECIAL Public/NRC/SCE Awareness Series by HAHN Baba Courtesy of The DAB Safety Team Causes of SONGS Unit 3 Replacement Steam Generators Fluid Elastic Instability Subject: Promoting Human Performance Tool “Critical Questioning & Investigative Attitude” SONGS Unit 3 FEI Root Cause: Lack of “Critical Questioning & Investigative Attitude” by SCE/MHI Design Causes: Narrow tube pitch/diameter ratio, too many tall tubes and lack-of in-plane supports Operational Causes: High Steam Flows, High Steam Velocities and Low Steam Pressures Primary Mechanistic Cause: Fluid Elastic Instability AKA Vapor Fraction >99.6% AKA Steam Dry-outs, Lack of Tube Damping (No Thin Water Film on Tube to Release Heat) Secondary Mechanistic Causes: Flow-induced Random Vibrations, Excessive Hydrodynamic Pressures (Mitsubishi Flowering effect) Human Performance Errors: 1. Lack of Solid Team Work, Alignment and Design Reviews 2. Avoidance of 10CFR 50.90 Amendment Process 3. Financial and Time Pressures 4. Lack of Academic and Industry Benchmarking 5.

Complacence and Negligence 6. Lack of NRC Region IV Strict Oversight 7. Lack of skills, experience and technology to design & build a CE Replacement Steam Generators

comment #64378 posted on 2013-03-01 02:42:04 by HelpAllHurtNeverBaba

February 26, 2013, 89.3 KPCC, Southern California Public Radio, "Boxer says documents from a whistle blower show SoCal Edison was trying to avoid having to reapply for a permit and was "aware" the repairs made to the plant aren't the ones that should have been done." I support the Honorable Senator Barbara Boxer 100%. I wish she had taken action earlier. 1. Dr. Pettigrew: So, you notice the U-bend -- the plane of the U-bend is being installed, and on top of the U-bends are bars. They are anti-vibration bars. And so you can see here that from the point of view of out-of-plane motion, the tubes are really very well supported because you have a large number of bars all around; but from the point of view of in-plane motion, there's really no positive restraint here to prevent the tube to move in the in-plane direction. Essentially, it relies on friction forces to limit the vibration. 2. Westinghouse states, "Test data shows that the onset of in-plane (IP) vibration requires much higher velocities than the onset of out-of-plane (OP) fluid-elastic excitation. Hence, a tube that may vibrate in-plane (IP) would definitely be unstable OP. A small AVB gap (3 Mil) that would be considered active in the OP mode would also be active in the IP mode because the small gap will prevent significant in-plane motion due to lack of clearance (gap) for the combined OP and IP motions. Thus, a contact force is not required to prevent significant IP motion. Manufacturing Considerations: None were extensively treated in the SCE root cause evaluation." 3. AREVA states, "At 100% power, the thermal-hydraulic conditions in the U-bend region of the SONGS replacement steam generators exceeded the past successful operational envelope for U-bend nuclear steam generators based on presently available data. The primary source of tube-to-AVB contact forces is the restraint provided by the retaining bars and bridges, reacting against the component dimensional dispersion of the tubes and AVBs. Contact forces are available for both cold and hot conditions. Contact forces significantly increase at normal operating temperature and pressure due to diametric expansion of the tubes and thermal growth of the AVBs. After fluid elastic instability develops, the amplitude of in-plane motion continuously increases and the forces needed to prevent in-plane motion at any given AVB location become relatively large. Hence shortly after instability occurs, U-bends begin to swing in Mode 1 and overcome hindrance at any AVB location. There are 36 U-bends in Unit 2 SG E-088 and 34 in SG E-089 with a separation less than or equal to 0.050 inches. These tubes are the first ones to break/rupture in 5 months or during an accident." 4. John Large States, "Causes of Tube and Restraint Component Motion and Wear: My study of the various OAs leads me to the following findings and opinion that; (i) degradation of the tube restraint localities (RBs, AVBs and TSPs) occurs in the absence of fluid elastic instability (FEI) activity; (ii) TTW, acknowledged to arise from in-plane FEI activity, generally occurs where the AVB restraint has deteriorated at one or more localities along the length of individual tubes; (iii) the number of tube wear sites or incidences for AVB/TSP locations outstrips the TTW wear site incidences in the tube free-span locations. I find that the 'zero-gap' AVB assembly, which features strongly in the onset of TTW, is clearly designed to cope only with out-of-plane tube motion since there is little designed-in resistance to movement in the in-plane direction - because of this, it is just chance (a combination of manufacturing variations, expansion and pressurization, etc) that determines the in-plane effectiveness of the AVB; (iv) Uniquely, the SONGS RSG fluid regimes are characterized by in-plane activity, which is quite contrary to experience of other SGs used in similar nuclear power plants in which out-of-plane fluid phenomena dominate. Moreover, from the remote probe inspections when the replacement steam generator (RSG) is cold and unpressurized, I consider it impossible to reliably predict the effectiveness of the many thousands of AVB contact points for when the tube bundle is in a hot, pressurized operational state., and (5) v) The combination of the omission of the in-plane AVB restraints, the unique in-plane activity levels of the SONGS RSGs, together the very demanding interpretation of the remote probe data from the cold and depressurized tube inspection, render forecasting the wear of the tubes and many thousands of restraint components when in hot and pressurized service very challenging indeed. 5. John Large continues, "Phasing of AVB-TSP Wear -v- TTW: I reason that, overall, the tube wear process comprises two distinct phases: First, the AVB (and TSP) -to-tube contact points wear with the result that whatever level of effectiveness is in play declines. Then, with the U-bend free-span sections increased by loss of intermediate AVB restraint(s), the individual tubes in the U-bend region are rendered very susceptible to FEI induced motion and TTW. Whereas the OAs commissioned by SCE broadly agree that the wear mechanics comprises two phases, there are strong differences over the cause of the first phase comprising in-plane AVB wear: AREVA claim this is caused by in-plane FEI whereas, the contrary, Mitsubishi (and Westinghouse) favor random perturbations in the fluid flow regime to be the tube motion excitation cause. Put simply: (i) if AREVA is correct then reducing the reactor power to 70% will eliminate FEI, AVB effectiveness will cease to decline further and TTW will be arrested; however, to the contrary, (ii) if Mitsubishi is right then, even at the 70% power level, the AVB restraint effectiveness will continue to decline thereby freeing up longer free-span tube sections that are more susceptible to TTW; or that (iii) the assertion of neither party is wholly or partly correct. As I have previously stated, I consider that AVB-to-tube wear is not wholly dependent upon FEI activity. 6. John Large continues, "Tube Wear Rates – Predicting the In-Service Period: SCE presents the findings of its commissioned OAs in a positive light, claiming that at 70% power the restarted Unit 2 plant will maintain RSG tube integrity for 16 to 18 months of continuous running, that is considerably longer than the proposed 150 day inspection interval. However, closer study of the OAs reveals that the reasoning behind important aspects of the deterioration period for the AVB effectiveness in Unit 2 is flawed, being overly dependent upon a number of uncertainties that I identify and expand upon in my affidavit. Some account of these uncertainties has been taken by AREVA in revising the TTW time-to-burst period down to 2.5 months which is well below the 150 days inspection interval but, without much justification, it determines and front-ends the time-to burst with a further 3.5 month AVB wear-in period, thereby delaying the onset of TTW and the unacceptable level of risk of tube burst to about 1 month longer than the proposed inspection period. I have little confidence in the outcome of the AREVA and other OAs projection of the time period through which the Unit 2 nuclear plant could be reliably expected to operate without a) incurring a tube failure or b) running at a greater risk of a tube failure occurring. This is primarily because (i) it is generally accepted that Unit 2 is following along the same path of deterioration as Unit 3 (AVB wear and loss of effectiveness preceding TTW), although the reasons why it lags so much behind are not at all understood by SCE and, indeed, subject to disagreement between the OA consultants; (ii) moreover, the pattern of AVB breakdown is not clear from the more advanced TTW degradation of Unit 3, thus the extrapolation to Unit 2 is not robust – again, there is disagreement between the OAs on this; so, it follows, (iii) there is very little justification in adding to the time-to-burst for Unit 2 tubes a 3.5 month AVB wear-in period, this is particularly so because so there is no certainty of just where Unit 2 is presently at along the path towards TTW wear. In account of these uncertainties, together with the uniqueness of the in-plane FEI in the SONGS RSGs that I will touch upon later, I consider that restarting Unit 2 to continuous running, even at 70%, will incur a great deal of change, test and experiment. 7. John Large continues, "Plugging of the at-risk tubes is not a satisfactory solution because it is the retainer bar that vibrates via random fluid flow processes at sub FEI critical velocity levels - these are likely to continue in play or, indeed, exacerbate at the proposed U2 restart at 70% power, leading to through-tube abrasion, the detachment of tube fragments, lodging at other unplugged and in-service tube localities, resulting in the so-called 'foreign object' tube wear." 8. Comments from Mel Silberberg [NRC-RES, Retired [Chief, Severe Accident Research Branch; Waste Management Branch] to Region IV: I am disappointed in the composition of the special panel! Where is the representation from NRC-RES? The issues at SONGS involve thermal hydraulics and material science. The NRC-RES and its contractors are experts in these areas. The Office of Research was created by the Congress for such situations. Two RES staff covering these disciplines and one or two consultants, serving as peer-reviewers. Perhaps there needs to be a separate peer review. Public confidence can only be gained using logical, informed measures as I described above. Inspection Reports are only one facet of the problem, no question. However, understanding the reasons for the fluid instability, possible cavitation corrosion effects, etc. are phenomena which require evaluation by T/H as well as materials experts, with appropriate oversight by the ACRS. The SCE, the nuclear industry, the NRC and the public need assurance, not educated guesses. I have not seen a bona fide attempt to understand resolve the issue such that all can be alert to potential problems. I still remain puzzled as to why the ACRS [at least one of the Subcommittees]. I am trying to reach the ACRS Exec. Director to discuss this point. Thank you." 9. According to NRC Insiders, "NRC does not really have experts in T-H, Materials and QA. You may find one or two at the ACRS and none at the ASLB." 10. ATHOS Modeling Limitations: NRC AIT Report states, "The result of the independent NRC thermal-hydraulic analysis indicated that differences in the actual operation between units and/or individual steam generators had an insignificant impact on the results and in fact, the team did not identify any changes in steam velocities or void fractions that could attribute to the differences in tube wear between the units or steam generators. The above analyses apply equally to Units 2 and 3, so it does not explain why the accelerated fluid-elastic instability wear damage was significantly greater in Unit 3 steam generators. The ATHOS thermal-hydraulic model predicts bulk fluid behavior based on first principals and empirical correlations and as a result it is not able to evaluate mechanical, fabrication, or structural material differences or other phenomena that may be unique to each steam generator. Therefore this analysis cannot account for these mechanical factors and differences which could very likely also be contributing to the tube degradation." Based on comments from Dr. Pettigrew and other researchers, the results of ATHOS Models for FEI are very time-consuming and expensive to conduct and the results can vary from 20 to 100%. Westinghouse states, "We've performed similar tests in the past, and that's what our analytical codes are based on, the data from the tests we've performed in the past. They may not be as extravagant as Dr. Pettigrew's, but yeah, we perform tests, and that's what our models are based on only out of plane. Basically, we're using the same tools that we've used. We're just staying within our comfort levels. We're not pushing our design limits. We're staying with what we know, what's been proven to work in the past." AREVA Staes, "And our analysis codes also are based upon testing that was performed in mockups and boilers in France, which is where most of the design work occurs for the replacements. But they -- those tests were performed in the late '80s and early '90s to validate some of the design changes that they were making to the components. In the same context, we are in France developing a new thermal hydraulic code. That's been underway, but as you might imagine, the development of a code that has to handle so many variables and these conditions that are very uncertain, it's time -- you know, you have to vet the process and make sure that, again, you're staying in the bounds of what you've known and your technology that you've used, and continually use that to benchmark anything new that you're working on or that you're developing. But as far as the pinnacle of the replacement market or the replacement design, I would say that most of it is fairly standard, you know, at this point. I don't think there's anything outside of the norm that anyone is looking at." John Large states, "The input energy is the dynamic velocity (~) of the two-phase fluid impinging on the tube. The energy dissipation is via damping which is strongly related to the two-phase mix of the fluid, here water and steam as described by the void fraction. Increase in steam content, a greater void fraction, reduces the damping and, correspondingly, the increased volume results in an increase of the impinging velocity. AREVA states, "At 100% power, the thermal-hydraulic conditions in the u-bend region of the SONGS replacement steam generators exceed the past successful operational envelope for U-bend nuclear steam generators based on presently available data." The inference here is that AREVA is comparing like-with-like, but that would require AREVA having undertaken an ATHOS flow analysis48 for each of the comparative SGs. This I consider unlikely because for this AREVA would have required access to very detailed information on the design geometry and flow paths throughout the comparative SG tube bundles – being a designer/manufacturer of steam generators itself, I very much doubt that AREVA would have had access to such proprietary information from competitor manufacturers. So since it is unlikely that AREVA would have carried out an ATHOS computer simulation for each of the five (A to F) comparative nuclear plants, then analysis is unlikely to be directly comparing two-phase fluid flow velocity distribution in the critical FEI regions of the SONGS and comparative plant SG tube bundles. I can only surmise that the analysis comparison is between the mean or average velocity within the overall tube bundle for SONGs and each of the comparative plants. Moreover, since the velocity distributions within each of the comparative plants, because of different design geometries, flow areas, etc, will not be identical, it is very unlikely that the mean or average velocity provides even a crude basis of comparison of the FEI potential of the SONGS RSGs." The question is as Arnie Gundersen asks, "How similar to the SONGS S/Gs are these other S/Gs? Do the other steam generators, for example, use alloy 670 tubes and have similar spacing, similar support structures, etc.? To the best of my knowledge and belief, no other steam generator in the nation is as large as those at San Onofre with broached tube supports, a tight Combustion Engineering tube pitch, and no stay cylinder. Therefore, comparing San Onofre to "several other successfully operating large S/G's" is simply not a valid engineering or scientific comparison." 11. Controversy regarding Removal of Central Stay Cylinder: Palo Verde and ANO Unit One 2 replaced the RSGs without removing the central stay cylinder, added more tubes and made the tubes taller (Read NUREG-1841) with NRC Region IV blessings and SONGS avoided that Blessing because it did not believe in the NRC Blessing Process in 2004 and 2013 as demonstrated in Yesterday's SCE/NRC Meeting. . NRC AIT Report states, "The licensee's bid specification required that the stay cylinder feature of the original steam generators be eliminated to maximize the number of tubes that could be installed in the replacement steam generators and to mitigate past problems with tube wear at tube supports caused by relatively cool water and high flow velocities in the central part of the tube bundle." Elimination of the stay cylinder to increase the added 377 tubes and increase the average length of the heated tubes for increasing the heat transfer area by ~ 11% in The RSGs caused the following problems: (a) John Large states, "Indeed, this need to increase the heat transfer area (ie putting more tubes into the RSGs) and, with this, reducing the steamside flow area, may have been a strong contributory factor to the enhanced FEI activity in the SONGS FSGs. Moreover, the location of the additional tubing, particularly in what I would describe as the lower swirl space immediately above the tube support sheet, may have contributed to and/or determined the unique in-plane flow characteristics of the SONGS RSGs." (b) Arnie Gundersen states, "The center section of the original San Onofre steam generators contained a key structural element called a "stay cylinder" and no steam generator tubes. In 2005 or early 2006, Edison made a management decision to eliminate this vital support pillar and add additional tubes in its place. In the original steam generator design, there was no heat input in this central area of the steam generator, because there were no tubes to add the heat. When Edison added almost 400 tubes (4% of the tubes) to the center of the tube bundle in the San Onofre Replacement Steam Generators, Edison effectively increased the power distribution to the center of the steam generator. This radical and unanalyzed design change moved 4% of the heat to the inside of the tube bundle while reducing the heat by 4% to the outside of the tube bundle. Adding this heat to the center of the bundle was then exacerbated by removing the egg crate tube supports and replacing them with a broached tube support plate design that further reduced flow to the center of the steam generator. As the NRC confirmed in its AIT report, a large steam void has developed near where the additional tubes were added in the Replacement Steam Generators (called fluid elastic instability) that allows many types of excess vibrations to occur. Fairewinds review of Figure 1 below from Edison's Condition Report clearly shows that the location within the steam generators where the steam "fluid elastic instability" has developed is precisely the region where the extra heat created by the 400 new tubes would create an excess of steam and various vibrational modes. While 4% may seem like a small change, it is not. Each San Onofre reactor generates a total thermal output of approximately 3400 megawatts of heat. If one mathematically converts 4% of 3400 megawatts of heat, it equals 135 megawatts, or to illustrate it differently: 180,000 horsepower of thermal heat that was transferred from the outside of the tube bundles to the center, (c) Unit 3 has historically produced more power than Unit 2 (1186 MWe vs. 1183 MWe, 1178 MWe vs. 1172 MWe). Westinghouse states, "In the U-bend region, the gap velocities are a strong function of power level. The steam flow in the bundle is cumulative and increases as a function of the power level and the bundle height which causes high fluid quality, void fraction, and secondary fluid velocities in the upper bundle." According to the Plant Procedures, Unit 3 RCS flow is 79.79 Million Ibs/hour and the delta T between Hot leg and Cold Leg is 58 degrees Fahrenheit. According to the SONGS Plant Procedures, Unit 2 RCS flow is 75.76 Million Ibs/hour and the delta T between Hot leg and Cold Leg is 57 degrees Fahrenheit. I go along with Westinghouse that higher power of 79.79 Million Ibs/hour caused FEI in Unit 3, because steam saturation temperature was achieved due to lower secondary side pressures of 833 psi and poor circulation ratios of 3.3 earlier in the U-tube bundle than anticipated due to increased average length of the tubes from 680 inches to 730 inches. The critical heat flux was achieved in Unit 3 area of wear (~ Hot Leg side Columns 75 -90, Rows 90-120, vertically located z-axis cut at about 20 inches above the 7th TSP) due to increased height of the bundle and narrow tube to pitch diameter. This caused high fluid quality, void fraction, and secondary fluid velocities in Unit 3 area of TTW. Unit 2 did not experience FEI, because power levels were low and critical heat flux was lower at the same point of wear as Unit 3, and by the time steam-water mixture achieved high fluid quality, void fraction, and secondary fluid velocities, it exited the u-tube bundle. It is also my observation, that FEI is an intermittent phenomena controlled by the varying circulation ratios and pressures in the steam generator. When this happens, the RCS return flow temperature in the cold leg is higher than usual, say by 4 degree Farenheit, even though the power is at a steady state level subject to CROSSFLOW uncertainty calculation of 0.5%. FEI causes movement of U-Tubes with large amplitudes starting from the tube sheet to the top of the U-bend and this could have conceivably caused tube-to-tube violent impact at the bottom of the tube sheet registering VLPMS alarms in Unit 3. Since Unit 2 did not experience FEI, no VLPMS alarms occurred in Unit 2. 12. Palo Verde RSG Design: The tube supports have three basic configurationsl-() horizontal grids (eggcrates/lattice) that provide support to the vertical run of the tubes, (2) vertical grids that provide vertical and horizontal support to the horizontal run of the tubes in the upper bend region, and (3) diagonal strips (batwings) that provide out-of-plane support to the 90-degree bends. The upper tube bundle support system (1) supports the horizontal tube spans against high velocity, two-phase cross flow, (2) permits an expanded vertical tube pitch (from 1.0 inch to 1.75 inches) so as to promote free flow through the bend region and prevent low-flow dryout regions, and (3) supports the upper tube bundle via structural beams against postulated accident condition loads, seismic loads, transportation loads, and dead weight. The U-bend support structure for the replacement steam generator differs from the original design in that it includes welded connections between the vertical grids and the diagonal (batwing) supports. Other features of the U-bend support system are that the batwings bisect the 90-degree bends, the bend region supports are perforated and narrower than the original design, and the bend region supports have ventilation holes. These changes in design improve the thermal/hydraulic conditions in the upper bundle region, preventing crevice dryout and reducing secondary-side fouling, as well as addressing tube-wear phenomena observed in the original steam generator. The diagonal strips (batwings) are located at every row and are designed to prevent out-of-plane deflection and thus preclude the deflection amplitude required for fatigue. The replacement steam generator design has an increased circulation ratio when compared to the original steam generator. 13. ANO Unit 2 RSG Design: Strict ovality control was implemented during the manufacture of the tubes to limit dimensional variability in the U-bend region. The thickness of the AVBs was also tightly controlled. To limit the potential for U-bend vibration and wear, AVBs support the U-bends. The AVBs provide sufficient support to the U-bend so that all the tubes remain elastically stable even if it is assumed that some of the support points are inactive. The AVBs in adjacent columns are inserted to different depths (i.e., staggered) to limit the U-bend pressure drop and to discourage the formation of flow stagnation regions. The AVBs are nearly perpendicular to the centerline of the tubes at all locations in the U-bend region to provide support without unnecessary tube contact. These features provide margin against flow stagnation, corrosion, and tube vibration. 14. The NRC AIT Report states, "The team identified that the design of the replacement steam generators did not expect any potential vibration concerns in the area of the tube bundle where the retainer bars were located. The basis for Mitsubishi's design philosophy relied on the following factors: (a) Based on the calculated natural frequency of the retainer bar, Mitsubishi considered that there would not be a resonant vibration condition relative to the flow conditions in the location of retainer bars, and (b) The vibration analysis of the tube bundle only considered out-of-plane vibration because in-plane vibration was not expected to be an operational concern for the retainer bars. The outermost tubes were considered the least susceptible to flow-elastic instability; therefore retainer bar locations were not included in the vibration analysis. Retainer Bars in other MHI SGs range with a frequency between 120-1180. Because of the excessive number of tubes due to change of Alloy 600MA to Alloy 690TT, the tube-to-tube clearance tightened towards the apex of the U-bend. Therefore, the restraint assemblies required a smaller diameter retainer bar with 56 HZ frequency in order to fit between the tube rows. 15. SCE SNO states: As a plant operator, I have operational control over two of the three components needed for in-plane fluid elastic instability. Specifically, reducing power can reduce steam velocity and reduce steam dryness sufficiently to preclude in-plane fluid elastic instability since the conditions for fluid elastic instability no longer occur concurrently. 16. SCE In

Enclosure 2 states, “A Probabilistic Risk Assessment (PRA) was performed to analyze the risk impact of the degraded SG tubes on SONGS Unit 3 SG 3E-088 with respect to two cases: (1) any increased likelihood of an independent SG tube rupture (SGTR) at normal operating differential pressure (NODP), or (2) due to a SGTR induced by an excess steam demand event, also referred to as a main steam line break (MSLB). The SONGS PRA model was used to calculate the increases in Core Damage Probability (CDP) and Large Early Release Probability (LERP) associated with each case. In both cases, all postulated core damage sequences are assumed to result in a large early release since the containment will be bypassed due to the SGTR; therefore, the calculated CDP and LERP are equal. The total Incremental LERP (ILERP) due to the degraded SG tubes (i.e., the sum of the two analyzed cases) was determined to be less than 2x10-7. This small increase in risk is attributed to two factors. First, the exposure time for the postulated increased independent SGTR initiating event frequency case was very short (0.1 Effective Full Power Month (EFPM)). Second, a MSLB alone does not generate sufficient differential pressure to cause tube rupture in Case 2. The differential pressure across the SG tubes necessary to cause a rupture will not occur if operators prevent RCS re-pressurization in accordance with Emergency Operating Instructions. 17. So from the above Information, I conclude that the “As-designed and Degraded Unit 2 RSG Tube-Bundle is not capable of preventing the FEI caused multiple tube ruptures from a MSLB with failure of a MSIV to close. We saw that in Unit 3 with failure of 8 tubes at MSLB test conditions. SCE SNO says, he has control of the plant operations, but SCE Enclosure 2 contradicts SNO with an “If statement” by stating,” The differential pressure across the SG tubes necessary to cause a rupture will not occur if operators prevent RCS re-pressurization in accordance with Emergency Operating Instructions.” 18. Operator Action as claimed by Edison to detect the leak by N-16 and other radiation monitors and the ability to re-pressurize the steam generators are not practical to stop a major nuclear accident in Unit 2 in progress in the first 15 minutes of a MSLB due to the following factors: • The operator action will not work due to the short duration of the initial and devastating event, the radiation/steam environment, communication errors between the control room and field operators due to sonic booms and hissing steam noises (sound-powered phones, pagers, cell phones and radios will not work in such an environment), darkness, difficult terrain and other unknown equipment failures/troubles [e.g. San Onofre’s auxiliary feed-water steam supply piping, which would provide water to the steam generators, if their main supply was lost is vulnerable to a big flood; A big fire in auxiliary feed-water pump room would knock 2 out of three pump’s electrical circuits, etc.], and other contingencies. • During a partial walk down of the Unit 2 high-pressure safety injection system in August 2011, NRC inspectors found a drain valve partially open, when it was required to be closed. “Operations personnel failed to implement instructions for filling, venting, draining, startup, shutdown, and changing modes of operation for emergency core cooling systems as written,” the NRC said. “seismic class I valves continue to be miss positioned, safety-related plant systems may be unable to accomplish their safety functions after an accident”. • One of the well-known SONGS Shift Managers told SCE Management that he was not going to put his “License on the line” by operating a “Defective Unit.” Several other shift managers have retired rather than work for SONGS’ Profit-Motivated and Retaliating Management. • The Operator Union has warned the SCE Management that with the proposed operator reductions, it will not be safe to restart Unit 2. • There have been 10 SGTRs (or significant leaks) in U.S. PWRs from 1975 to 2000. Human performance weaknesses, such as misdiagnosis, substantial delays in isolating the faulted steam generator, and delayed initiation of the residual heat removal system, have been identified in these events. The events also involved unnecessary radiation releases; lack of RCS subcooled margin, excessive RCS cooldown rates, and overfilling the SG because of human or procedural problems. • Additional complications would add to operator burdens. These include high noise levels preventing normal communications; RCS cooldown with potential recriticality; actions to recover RWST inventory; many radiation alarms, unexpected high radiation areas in the turbine building, and atmospheric releases; fire alarms and fires from steam and shrapnel from the break; and emergency communications with local, state, and Federal governments diverting operations personnel before the technical support center is manned or additional operations personnel arrive on site. The Halden Control Room Staffing study found poor operator performance in one of two simulations of a SG leak with a failed open SG safety relief valve, as well as simulations where crew size was decreased to attend to other duties. • Below are some of the weaknesses witnessed during review and/or observation of the Simulator Evaluations, Emergency Planning Drills and discussions with the Shift Managers during 2012. Each weakness may be attributed to one or the other Drills/Exercise Performance (DEP) Miss-classifications: A. Unclear and confusing Emergency Action Levels (EALs) and less than adequate Basis Documents. B. Too many Priority Reading Assignments to clarify the EALs and Basis Document. C. Lack of solid teamwork between the Operating Crew, Control Room Supervisor (CRS), Station Technical Advisor (STA) and Emergency Coordinator/Shift Manager (EC/SM). D. Crew members confused and concerned about their roles and responsibilities. Crewmembers held back or failed to provide information, which resulted in SM and CRS to trip the reactor. E. Poor communications between the Operating Crew, CRS, STA and EC. Briefs were ineffective at focusing on the crew priorities. Three way communication not used for direction or when providing information relative to plant status. F. Poor diagnostics/interpretation of the transient events by the Operating Crew, CRS, STA and EC. Serious omissions, delays, or errors made in interpreting indications resulting in degraded plant conditions. Failed to use, or misused, or misinterpreted indications that resulted in improper diagnosis. G. Procedures were not followed correctly which impeded plant recovery or caused unnecessary degradation of plant conditions. Crews did not recognize EOI Entry Conditions. H. Repeat failures of the STA to provide consistent & independent check of the EAL by EC. I. Lack of Stringent Operations Department/NTD Evaluation and Remediation Criteria for SM/STA/ Operations Crew to achieve excellence and eliminate above shortcomings to prevent DEP Failures. J. Lack of practice by the Operating Crews, CRS, STA and EC following the coaching/critique provided by the OPS SM Supervisor and NTD Evaluators. • During the April, 2012 Fire Notification of Unusual Event, it took 40 minutes between the Control Room and Electricians to find the drawings to determine the location of the breaker to de-energize the power to the electrical panel in the Unit 2 turbine building to extinguish the fire and terminate the event. Luckily, the Unit 2 was in shutdown and the SONGS Fire Department was present at the scene to extinguish the fire, if it got out of the control. Later it was determined, that the Fire Department and Control room did not take timely action to extinguish the fire due to an over-conservative fire procedure. • In 2001, Ratepayers lost 100 million dollars in a Unit 3 Switchgear fire due to faulty alarms and miscommunication between the SONGS Fire Department and Control Room. Unit 3 was in shutdown for 5 months due to Main Turbine repairs, which was damaged in the fire event. • During the 2011 NRC/FFEMA Evaluated Exercise, the General Emergency Declaration was missed by 29 Minutes due to a communication error between the Emergency Offsite Facility Health Physics Supervisor and Technical Support Center Station Emergency Director. If this was a real event, the public would have been potentially subject to offsite radiation releases unnecessarily for 29 minutes. THANKS To NRC FOR POSTING THIS BLOG

NRC Hosts Webinar on Palisades Leaks

posted on Tue, 22 Jan 2013 14:15:39 +0000

Viktoria Mitlyng
Senior Public Affairs Officer
Region III

We gathered at the NRC's Region III office near Chicago on a recent Saturday morning to continue our dialogue with the public about the Palisades nuclear plant. We decided to host our second webinar on this plant on a Saturday in response to a request from members of the public to hold it at a time when people aren't at work. Close to 100 people listened to the NRC's presentation by four representatives of the Region III staff and asked questions on a wide range of questions on recent problems at Palisades. The purpose of the webinar was to talk about the NRC's regulations on a specific category of leaks – including the leaks that occurred at Palisades in 2012 – and the NRC's response to these leaks. They are called "through-wall" leaks because they come through the wall of pipes and other plant components important to safety. Resident inspectors stationed at every nuclear plant in the country continuously monitor any such leaks making sure they are properly understood and handled. Leaks that have no safety impact are not regulated by the NRC. NRC's regulations on through-wall leaks are based on the safety significance of the affected equipment. Leaks from the pressure retaining boundary of the reactor coolant system are not allowed and must be fixed right away. Other types of leaks may not require immediate repair but must be fixed before they have a negative impact on plant safety. We talked about four through-wall leaks identified at Palisades last year; one of these was discovered by an NRC Resident Inspector during a routine daily inspection. Even though these leaks did not compromise plant safety, they concerned us because of their frequency. The agency decided to commit additional resources this year to evaluate these leaks and determine whether they represent a weakness in the plant's maintenance program. Three of the four leaks at Palisades have been fixed. The remaining leak from a refueling water tank is closely monitored and will be repaired according to NRC regulations. We informed the public when the leaks at Palisades were discovered even though the NRC doesn't normally make public notifications on leaks of very small safety significance. This was done in response to requests from many people to be informed about such issues at the plant. We will continue the high level of engagement with the public near the Palisades plant to meet the agency's goal of openness and transparency. Additional webinars on reactor vessel head embrittlement and environmental monitoring are already in the works. In addition, the NRC staff will have a booth at the Garden and Leisure Show in Benton Harbor, Mich., March 15-17.

Comments

Fort Calhoun Nuclear Plant – A 2013 Update

posted on Thu, 24 Jan 2013 16:06:59 +0000

Lara Uselding
Public Affairs Officer
Region IV

As we turn the page on a new year, the NRC is watching closely as the operators of the Fort Calhoun nuclear plant, located in Omaha, Neb., are working around the clock in hopes of returning the plant to service. It remains to be seen if the NRC is convinced the efforts of the Omaha Public Power District's (OPPD) are sufficient. The plant has been powered down since April 9, 2011, for a refueling outage. The outage was extended due to historic flooding along the Missouri River followed by an electrical fire that led to an "Alert" declaration and further restart complications. On Jan. 8, OPPD officials and the NRC Fort Calhoun Oversight Panel members met before the five-member NRC Commission to discuss the current plant status. Positive change is on the horizon. "They [OPPD] are looking at problems with a different set of eyes today," said Mike Hay, NRC Branch Chief and panel member. Some NRC Commissioners also noted the efforts by OPPD management to turn things around. It is also clear more work needs to be done. In November 2012, the NRC issued a detailed inspection plan listing some 450 items that require attention, inspection, and resolution. Many of these items are subsets of the familiar issues that have been reported over the past two years including the breaker fire, flood strategy concerns, containment penetrations, and containment internal structures issues. In 2013, there will be numerous NRC inspectors carrying out a very rigorous inspection schedule. A five-member team has already been on site for two weeks to independently verify results from a third-party safety culture assessment done last year. As part of the inspection, NRC held focus group interviews with plant works to assess the current climate and help the NRC understand how in tune management is with staff. Later in February, these results will be used to fuel a second, larger team inspection to fully assess human performance and safety culture at Fort Calhoun. There is more to come. There will be an announcement soon with details for the next public meeting in Nebraska. The staff will continue to post updates and helpful information to the Fort Calhoun specific Web site.

Comments

comment #49237 posted on 2013-01-30 11:12:46 by SafetyAdvocate

Re: second PR Answer to Q4: Again, the question is, in light of the discovery of inadequate embedment for j-bolt anchors for all four raw water pumps, will the NRC insist that all anchors at Fort Calhoun be examined. The NRC's answer in this blog has twice failed to answer this question. It appears the NRC wants to dodge this question.

comment #46292 posted on 2013-01-24 12:38:46 by CaptD

RE: NRC held focus group interviews with plant works to assess the current climate and help the NRC understand how in tune management is with staff. If Ft. Calhoun is anything like San Onofre then everybody interviewed knows their job is on the line and therefore watches what they say, so the amount of candid information received is almost nil.

comment #47048 posted on 2013-01-26 10:10:44 by SafetyAdvocate

Re: PR Answer to Q1: The NRC is moving far too slowly to deal with the known high-risk flooding hazard currently present at Fort Calhoun. As a part of the post-Fukushima NRC actions, NRC isn't requiring Fort Calhoun to complete its evaluation of flooding hazards from upstream dam failures until 2014. NRC's handling of generic issues takes years, sometimes decades. The 0350 panel already has the data it needs to act now by not allowing Fort Calhoun to restart until adequate measures are taken to protect from the known high risks from flooding caused by upstream dam failure. Many Nebraska and Iowa residents have little, if any, confidence in the NRC's track record re: flooding risks at nuclear plants or in the Corps of Engineers' management of the six dams on the Missouri River upstream of Fort Calhoun. The NRC needs to get real about the current known high-risk flooding hazard at Fort Calhoun. Re: PR Answer to Q3: The question was, will the NRC require Fort Calhoun to reconstitute all its design-basis documents prior to considering restart. This was not answered. Re: PR Answer to Q4: Fort Calhoun's Current Event Notification Report for Dec. 3, 2012 states, "Existing analysis requires a minimum embedment of 60 inches for a j-bolt type anchor." http://www.nrc.gov/reading-rm/doc-collections/event-status/event/2012/20121203en.html What was recently found at Fort Calhoun was only 9-inch embedment for j-bolt anchors for all four raw water pumps, rendering them all inoperable. The question was, in light of this discovery, will the NRC insist that all anchors at Fort Calhoun be examined. This was not answered.

comment #49382 posted on 2013-01-30 14:38:33 by Moderator in response to comment #49237

Yes, the inspectors who arrived on site Monday to look at the cooling water pump bolts will be inspecting to verify if similar anchor bolt problems exist elsewhere at the site. The press release announcing the inspection team's arrival at the plant can be found here: http://pbadupws.nrc.gov/docs/ML1302/ML13028A210.pdf Lara Uselding

comment #46780 posted on 2013-01-25 15:07:05 by Moderator in response to comment #46667

ANSWER TO Q1 : The NRC's recognizes the need to address the cascading dam failure flooding issue at nuclear power plants and it is currently being addressed by two agency organization activities: 1) the Japan lessons Learned Directorate (JLD) charged the NRC staff to address the Fukushima near-Term Task Force (NTTF) Recommendations that address flooding and ; 2) The Research arm of the NRC is looking at this issue under its Generic Issue-204 research entitled "Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure." ANSWER TO Q2: In 1985, the NRC issued generic communication to licensees on the use of Teflon material in containment piping. OPPD did replace penetrations with Teflon in cables connected to safety related loads and instrumentation. However, there are penetrations that contain Teflon that were not replaced because they did not contain safety related loads or instrumentation. This issue is currently being looked at by the increased oversight panel known as the 0350 panel and will need to be addressed prior to restart. ANSWER TO Q3: NRC regulations require that plants have updated documentation for structures, systems and components. If design basis documents are missing, the NRC will require the plant operator to reevaluate and reproduce them. ANSWER TO Q4: The pumps are required to have 16 inch bolt anchors, not 60 inch. The plant operator has replaced the anchors and NRC will be inspecting their work. I hope this information is helpful. Lara Uselding

comment #46779 posted on 2013-01-25 15:04:05 by Moderator in response to comment #46264

ANSWERS FOR Q1 AND Q2 NRC does not require a plant to redo the original plant design. NRC regulations require that plants have updated documentation for structures, systems and components. If design basis documents are missing, the NRC will require the plant operator to reevaluate and reproduce them. The design basis document and internal structures issues are currently being evaluated and inspected under the NRC's increased oversight process and must be resolved by the licensee. ANSWER for Q3 This matter has been discussed at previous public meetings. All videos from previous meetings including those in which NRC staff have discussed this can be found at http://www.nrc.gov/info-finder/reactor/fcs/special-oversight.html ANSWER for Q4 The maintenance building column has been fixed and the other is in process. ANSWER for Q5 To see the comprehensive 450+ items outlined in the NRC basis document that need to be addressed and resolved by the licensee and inspected by the NRC please visit this link: http://pbadupws.nrc.gov/docs/ML1231/ML12318A319.pdf I hope this information is helpful. Lara Uselding

comment #46289 posted on 2013-01-24 12:33:27 by CaptD in response to comment #46264

Great comment Salute!

comment #46258 posted on 2013-01-24 11:50:15 by renodeano

Brings back memories of their poor performance in the early 80's and the get well program for them.

comment #46264 posted on 2013-01-24 12:02:16 by LillyMunster

Is the NRC going to answer the more important questions the public has been asking about Ft. Calhoun? 1. Will the plant be required to reconstruct their plant design documents since so many are missing or incorrect? They couldn't even find underground pipes under the turbine building properly because the design documents are so messed up. There are also many problems with the containment design documents. They just found out now that internal containment structures are improperly designed and unsafe for operation. What else involved with the containment structures or the plant are wrong or misunderstood because the design documents are wrong or missing? 2. Could the NRC explain why they have known since the 1980's that Calhoun's design documents were incorrect or missing yet never forced Calhoun to fix this even when they gave a license extension to run another 20 years? 3. Will the NRC publicly address the issues of under plant erosion? The first geo-testing found considerable erosion and building failure problems under the turbine building & machine building and the engineering firm thinks there may be the same under the containment building and the building housing the spent fuel pool. Why is this not being conveyed clearly to the public. This is a much bigger deal than the human resources issues at the plant. Has this additional 3rd party inspection work been done? Knowing if there is erosion or other problems under containment and the fuel pool is a big deal, why is this never talked about? 4. Why did the NRC never make Calhoun fix this under plant erosion problem even though they have known about it since the previous flood in 1993? It has obviously gotten worse over the years including a sinking column in the maintenance building and voids under the turbine building. The erosion seems to be in the backfill layer under the plant but it doesn't mean it is not a problem. 5. Could the NRC tell the public clearly what is being asked of Calhoun to fix these underground erosion problems, the lack of required accurate design documents and the unknown erosion problems under containment & the spent fuel pool? Will they be required to fix any of these before they are allowed to restart and if so what will be required as far as repair work? It is really worrying that the NRC seems focused on human resources issues and is not telling the public anything about these rather major problems at Calhoun that absolutely should be part of the public disclosure on the plant.

comment #47311 posted on 2013-01-26 23:05:46 by LaVerne Thraen in response to comment #46780

The two agency flood related activities won't be complete in time for Fort Calhoun restart. We have been told by the NRC that the restart will not be held up waiting for those reports. Lara all penetrations through the containment building are safety related as made clear by Commissioner Apostolakis at the last teleconference. Your just repeating OPPD's lame excuse for not changing them in the 1990's. If it was true that the NRC needed fully reconstituted design basis documents to operate they would not have been operating for the last 20 years. 1989 was the first notice from the NRC about the lack of design basis documents.

comment #47316 posted on 2013-01-26 23:10:58 by LaVerne Thraen in response to comment #46779

So the Maintenance Building column is fixed and the other is in process. Which column is the other that is being fixed? The second column which one is that?

comment #47815 posted on 2013-01-28 01:13:02 by James Greenidge

Let's fix it right and roll it out, NRC! James Greenidge Queens NY

comment #48796 posted on 2013-01-29 19:26:37 by Mary Ann Krzemien

Fort Calhoun's failure to control and maintain its design-basis documents is a big no-no. The lack of documentation that shows all calculations has implications throughout the plant. Systems, structures, and components can't be properly evaluated or modified without them. OPPD's recent revelation to the NRC that the beams and columns inside Fort Calhoun's containment structure weren't constructed as design drawings specified and that calculations were missing is one very serious example of OPPD's lack of control over its design configuration. Fort Calhoun's anchor embedments that don't match design drawings may be another example of this problem with critical ramifications to the entire plant. The NRC has been asleep at the switch for a long time regarding these problems at Fort Calhoun.The 0350 panel needs, at long last, to get an adequate grip and require OPPD fix all of these serious problems before restart is approved. The panel has plenty of authority to do this. A panel member explained in a public meeting last summer that the 0350 panel can specially design its enforcement to deal with all the problems existing at Fort Calhoun prior to restart.

comment #48507 posted on 2013-01-29 09:47:13 by Moderator in response to comment #46779

Yes, the maintenance building column is being fixed and the voids under the turbine building (not a second column) are being addressed. Lara Uselding

comment #48519 posted on 2013-01-29 10:18:18 by Moderator in response to comment #47048

In regards to Q3: A decision has not been made on this issue. It is an item on the restart checklist (the document NRC is using to inspect the plant), under the section entitled engineering design and configuration control, and it has not been inspected yet. Lara Uselding

comment #48523 posted on 2013-01-29 10:22:12 by Moderator in response to comment #47048

RE: Q4 The NRC launched a series of inspections at the Fort Calhoun nuclear plant yesterday to examine the plant's restart checklist and evaluate a recent cooling water pump issue. NRC inspectors will evaluate a condition, reported on Dec. 2, 2012, involving four main water pumps incorrectly anchored using shorter bolts than required. Shorter bolts may have caused the pumps to be inoperable following extreme ground motion during an earthquake. Lara Uselding

comment #46644 posted on 2013-01-25 08:23:39 by Moderator

Note from the Moderator: A three-part series of comments about the San Onofre plant from HelpAllHurtNeverBaba helpallcqiascnp@yahoo.com has been posted to the Open Forum section of the blog. You can find them here: http://public-blog.nrc-gateway.gov/2012/08/01/an-open-forum-now-available/#respond

comment #63526 posted on 2013-02-27 09:21:51 by in response to comment #62869

WOULD YOU FEEL SAFE ENOUGH TO LIVE CLOSE TO PLANT WITH YOUR CHILDREN PLAYING IN YARD?OR ANYWHERE ELSE IN THAT AREA?????

comment #46667 posted on 2013-01-25 09:39:12 by SafetyAdvocate

Lilly's questions and comments are right on target. I know of many people in Nebraska and Iowa who are asking the same questions. The NRC needs to give clear, on-point answers to each. The NRC's publicized focus on human resources issues while giving short shrift to the many very significant technical problems at Fort Calhoun is a failure to protect the public adequately. Here's some additional questions the NRC needs to answer: 1. When will the NRC deal realistically with the ongoing documented high-risk flooding threat at Fort Calhoun from upstream dam failures? It appears that both Fort Calhoun and the NRC are relying solely upon Army Corps of Engineers predictions and management of the six earthen dams on the Missouri River upstream of Fort Calhoun. This is foolhardy and unsafe given how outmatched the Corps was regarding the 2011 flooding. 2. Why did the NRC allow Fort Calhoun to continue to use Teflon electrical penetration seals after the mid-1980s even though it was then known that they degrade when encountering high radiation? Fort Calhoun is apparently the only nuclear plant that continues to use Teflon. This fact exemplifies Fort Calhoun's poor decision-making re: nuclear safety. Fort Calhoun's seals which were extracted and recently lab tested are reported to have greatly deteriorated. Why hasn't the NRC stepped in and assured safety when Fort Calhoun has so clearly hasn't? 3. Many serious problems exist at Fort Calhoun apparently plant-wide, including failure to maintain and update design-basis documentation as required. This is despite the NRC's explicit alerts in 1992 and 1996 to licensees that this documentation, including calculations, should be maintained. Will the NRC require Fort Calhoun to reconstitute all its design-basis documents prior to considering restart? 4. Another potentially plant-wide problem is inadequate anchor embedment. Existing analysis requires a minimum embedment of 60 inches for a j-bolt type anchor but only 9-inch embedment has been found for all four raw water pumps. In light of this discovery, will the NRC insist that all anchors at Fort Calhoun be examined?

comment #46672 posted on 2013-01-25 09:51:15 by Nancy in response to comment #46264

I agree. Will the NRC please answer these questions?

comment #48214 posted on 2013-01-28 17:38:20 by Hiddencamper in response to comment #48124

All that matters is the plant design is adequate to ensure safety of the plant. This could mean waterproof pump house. This could mean flood barriers. This could mean portable pumps which could be hooked up well in advance to any flood hitting the site. There are multiple ways to ensure that critical safety functions are accomplished.

comment #48215 posted on 2013-01-28 17:49:27 by Hiddencamper in response to comment #47048

All plants have to complete post fukushima actions regardless of whatever other commitments they have. Fort Calhoun was not shut down because of flood related issues, so there is no reason they have some arbitrarily earlier deadline placed on them than the rest of the industry to complete. Additionally, inspections like this at nuclear plants take a long time and require a lot of resources. Trying to crunch that time will just result in them, and every other plant, not having an adequate evaluation of flooding hazards, and I think all parties would rather have a good evaluation than a half assed one released "on time". Most things in nuclear take years to change or evaluate. With regards to design basis documents I think the NRC was very clear, Fort Calhoun will need to have updated design basis information. If the NRC finds they do not have up to date design basis information sufficient to demonstrate plant safety, they will be forced to update it. I think the question you should be asking is: "Does Fort Calhoun currently have adequate updated documentation for all of its SSCs, and if they do not, will they be required to produce it prior to startup".

comment #48198 posted on 2013-01-28 16:39:57 by Reno Deano in response to comment #48124

> Evidently you do not understand the professionalism of the NRC staff. Equating them to a "Fox in a hen house" is very disingenuous to the public's nuclear safety inspectors.

comment #48124 posted on 2013-01-28 13:25:40 by american2018@gmail.com in response to comment #47815

> So the fox is going to do a real good inspection of the hen house. Right. I have a question, were those cooling water pumps submersible or not? If you jokers destroy the Corn Belt, there's not going to be any place in America for you.

comment #62869 posted on 2013-02-26 08:43:38 by SafetyAdvocate

> The NRC is holding a closed-door meeting today at NRC headquarters in Rockville on Fort Calhoun's security. The agenda refers to an Exelon security assessment of FCS and a security plan. This strongly suggests that the security deficiencies at Fort Calhoun are significant.

NRC Joins Five Other Agencies in Reporting on Navajo Land Contamination

posted on Mon, 28 Jan 2013 17:50:08 +0000

Maureen Conley
Public Affairs Officer

The government has made good progress in reducing risks from uranium contamination on Navajo land, five federal agencies told Congress in a report last week. EPA compiled the report with input from the NRC, the Department of Energy, the Bureau of Indian Affairs, the Centers for Disease Control and the Indian Health Service. This report recaps work done since October 2007. At that time, Congress asked the agencies to develop a five-year plan to address the contamination, which dates back to the 1940s. Demand for uranium skyrocketed near the end of World War II. The ore was needed for nuclear weapons manufacturing and later to fuel commercial power reactors. The Navajo Nation lands had large uranium deposits, but mining and milling then was not nearly as regulated as it is today. Mining companies left extensive contamination requiring cleanup. In 1978 Congress passed a law to ensure that uranium mill waste (called tailings) would be safely managed into the future. Under that law, DOE is responsible for the long-term care and maintenance of four former mill sites: Tuba City, Ariz.; Shiprock, N.M.; Mexican Hat, Utah; and Monument Valley, Ariz. The NRC oversees DOE's work at those sites. For example, DOE is responsible for cleaning up contaminated groundwater at the sites. The NRC reviews those cleanup plans. DOE monitors disposal facilities for uranium mill tailings. The NRC observes DOE inspections at the sites. The NRC also reviews and comments on DOE's performance and environmental reports. While the NRC does not regulate mine cleanup, the agency will also be working closely with EPA, DOE, the New Mexico Environment Department, and the Navajo Nation during the cleanup of a contaminated mine site in Church Rock, N.M. This conventional strip mine operated from 1967 to 1982. EPA plans call for the mine waste to be disposed at the nearby Church Rock mill site, which must be done in compliance with NRC disposal regulations. Over the past five years, NRC staff has met many times with members of the Navajo Nation. We will continue these oversight and outreach activities.

Comments

comment #56418 posted on 2013-02-15 18:43:02 by Disney

> Im attempting to sign up for the NRC (New California Republic) Faction but dont know how ive been almost everywhere but i can do jobs for them for caps i essentially want to be part of them can a person tell me how ?

comment #53354 posted on 2013-02-06 23:47:04 by ssentherbalshop

> WE SHOULD JUST BAN THE TRADE IN URANIUM BY ALL, SO THAT WE DONT HAVE A NUCLEAR CONFRONTATION IN THE FUTURE.

comment #53355 posted on 2013-02-06 23:48:01 by ssentherbalshop

> less pollution is always a good thing

comment #53356 posted on 2013-02-06 23:49:34 by bk

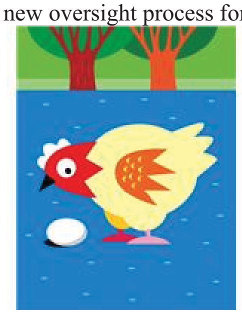
i dont like anything nuclear, the cost is just too much, the cost on humanity

Construction Oversight Pilot Builds on Agency's Longstanding Reactor Oversight Process

posted on Wed, 30 Jan 2013 18:35:03 +0000

Joey Ledford
Public Affairs Officer, Region II

The NRC is piloting a new oversight process for nuclear units under construction that is reminiscent of the old riddle, "What came first, the chicken or the egg?"



Obviously, the Reactor Oversight Process, or ROP, has been in effect for years. The NRC staff recently developed a Reactor Oversight Process for construction, known as cROP, designed to inform oversight of the ongoing work at Southern Nuclear Co.'s two new units at Plant Vogtle near Augusta, Ga., and SCE&G's two new units at V.C. Summer near Columbia, S.C. The new process uses numerous features of the original ROP, including the inspection program, assessment process and enforcement policy. But the construction ROP has its own Action Matrix and employs a construction significance determination process to assess the importance of inspection findings. Senior officials in the Office of New Reactors, or NRO, held public meetings near both sites this month to explain the program and gather public comments on possible revisions to improve it. Another public meeting is scheduled for Feb. 6 at NRC headquarters to evaluate the pilot, with the goal to report to the Commission by the end of April. The inspection program is a joint effort of Region II and NRO. Three construction resident inspectors are at each site, supplemented by regional specialists in various disciplines ranging from welding to concrete. Inspectors from headquarters monitor and review the performance of suppliers who ship safety-related components to the sites. The NRC estimates that the agency's inspectors will perform some 30,000 hours of inspections for each new unit before the process ends. Specifically, the inspection regimen requires the licensees to verify they have met 875 different ITAAC, or Inspections, Tests, Analyses and Acceptance Criteria. This comprehensive oversight program means any unit that is built would be constructed according to all applicable NRC regulations.

Comments

comment #49854 posted on 2013-01-31 09:23:17 by Moderator

For information on job opportunities at the NRC, please go to this page: http://www.nrc.gov/about-nrc/employment.html Moderator

comment #49421 posted on 2013-01-30 16:07:27 by Marvin L. Curland, (P.E. Ret.)

Please consider using personnel who have past experience in the business of design, construction, and testing of nuclear powered Naval nuclear vessels. They may be retired or available from current employment to provide considerable and useful assistance in Construction Oversight activity. I am retired from years of experience including 24 on nuclear submarines (Nautilus to Trident) in engineering, project management, nuclear quality assurance, and Chair of Joint Test Group for S5W Boats at Electric Boat/Gen. Dynamics and the remaining years in the commercial power generation area including on-site new construction and nuclear supplier activity. I retired formally November 1211; I can provide references and am willing to help at little or no fee plus expenses. This is no joke, please contact me.

Getting All the NRC News That's Fit to Send (To Your E-Mailbox)

posted on Fri, 01 Feb 2013 15:09:06 +0000

Eliot Brenner
Public Affairs Director

There is no shortage of ways to keep up with what the NRC is doing. Certainly, this blog is one good way and subscribing to our Twitter feed – which includes all notices of new blog posts, press releases, YouTube videos and more – is ideal. But there are other ways as well. The NRC maintains a number of specific lists to which you can subscribe, including: • ADAMS User Group • Generic Communications • Generic License Renewal Correspondence • Inspector General Reports • National Source Tracking System: Blog • New Rulemaking Dockets • News Releases • Open Government • Operating Reactor Correspondence • Part 21 - Report of Defects and Noncompliance • Reactor License Renewal • ROP Performance Indicators

- Approved FAQs • Speeches There is no charge for subscribing to any of the lists and you can unsubscribe at any time. You can sign up here.

Comments

The Public Meeting on Public Participation That Wasn't All That Public

posted on Thu, 31 Jan 2013 19:52:58 +0000

Darren Ash
Deputy Executive Director for Corporate Management

The irony was not lost on us when we were told the live webcast feed for today's Commission meeting on public participation wasn't, well, actually getting out to the public. Obviously, we encountered technical difficulties. The contractor hired to broadcast the meeting discovered early in the meeting that the webcast video was not being distributed to internet viewers (for reasons not yet clear). Unfortunately, it was not a quick fix to get it back up and running, and only the last part of the meeting actually ended up being available via live webcast. We apologize for this unfortunate turn of events. The archive of the meeting will be available in the archive section of the Webcast Portal later today.
Note: The webcast is now available.

Comments

comment #50179 posted on 2013-01-31 16:40:54 by LillyMunster

> Is something going to be done to deal with this? Since it was supposed to be a public engagement type discussion rather than public access to watch a meeting it seems a bit pointless if people couldn't participate? I know there was a problem on a web meeting a month or two ago where some of the participants could not be heard while others were quite loud. There was a phone bridge only on Ft. Calhoun recently that had a participant that kept yelling into the open phone line and playing music so the meeting could not be heard. Not sure if there is a way to boot someone out of a phone meeting but it would have been quite handy if someone could have identified and booted that participant. In their defense I think they were having a technical problem but it still disrupted the meeting. Maybe other technology platforms could be considered for these two meeting formats? It seems like the web meetings frequently have bugs in the system, or at least the ones I have been on.

comment #50344 posted on 2013-01-31 23:39:30 by HelpAllHurtNeverBaba

> Respected Mr. Darren Ash NRC Deputy Executive Director for Corporate Management Dear Sir, Admitting mistakes or difficulties (human performance or technical) and taking proven and systematic actions to correct them and continued monitoring and awareness are the biggest quality of a person, organization, nuclear utility and a nation. Finally, what I mean is, Americans need safe, affordable, reliable, well managed, well maintained and excellently operated nuclear power plants, where workers are free for raising nuclear safety and personnel concerns and Rate Payers, Regulators, Politicians and News Media are Proud. Thanks for posting this blog ... HAHN Baba

comment #50116 posted on 2013-01-31 15:07:00 by CaptD

> Please reschedule the meeting so that it is actually a public meeting and allow ample time for public comments since that is what the meeting is about! + Also consider giving a cash award to the top 5 suggestions, if you can pay Staff to attend meetings the NRC should reward those that make the NRC more efficient! A Win-Win for US... + I would encourage readers to read the comments on this NRC blog: "Improving Communication at the NRC" http://public-blog.nrc-gateway.gov/2013/01/16/improving-communication-at-the-nrc/

comment #54609 posted on 2013-02-10 13:08:00 by Jane Swanson

> I obtained the meeting transcript. I do not appreciate the remark (on page 30 or 32 I think) about the folks in California being asleep. I was one of the members of Mothers for Peace trying to access the live stream. I phoned several phone numbers on NRC website to ask if the problem originated with NRC or if I was doing somethong wrong. All I got were answering machines. I have not seen any evidence of the NRC - or AEC - valuing public participation in the 40 years MFP has had intervenor status. This recent failure is just one more example of the lack of respect and the roadblocks to public participation that are embedded in the culture of this captured regulator.

comment #50697 posted on 2013-02-01 10:20:23 by Moderator in response to comment #50179

Note from the Moderator: The meeting held yesterday on public participation was a formal meeting between the Commission and its invited participants. The public was invited to view in person or via webcast, but per Commission policy, there wasn't an opportunity for public comment during the meeting. However, anyone can submit their comments directly to the Secretary of the Commission by sending them to NRCExecSec@nrc.gov. They will be included as part of the record of the meeting. More information on Commission meeting policy is here: http://www.nrc.gov/about-nrc/policy-making/internal.html

comment #50699 posted on 2013-02-01 10:23:58 by Moderator in response to comment #50179

Thank you for your observations on our recent technology issues. We are still reviewing what happened yesterday and hope to take action that will minimize such issues in the future. I can't speak to the Ft. Calhoun phone bridge issue, but will make sure Region IV sees this feedback. Darren Ash

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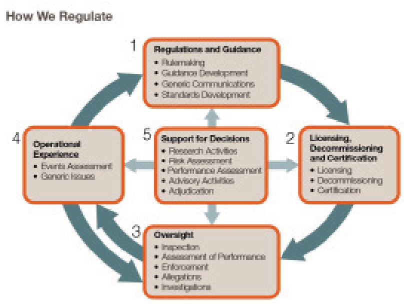
Stakeholder Input to Revised Enforcement Policy – Value Added!

posted on Tue, 05 Feb 2013 18:27:03 +0000



Lauren Casey

Enforcement Specialist
Office of Enforcement

The NRC establishes regulations, inspects those we license to make sure they are meeting those regulations and, at times, steps in to enforce regulations that licensees are violating. This last activity – enforcement – is a very important part of NRC's oversight. Key elements of the enforcement process are transparency and openness. Because of that, when the NRC determined the Enforcement Policy should be revised to reflect changes in our regulatory environment and the way the NRC and our stakeholders do business, we made sure to get input from you. For example, we held public meetings and asked for public comments in the Federal Register on the proposed revision. With the information from this outreach in hand, the NRC revised its Enforcement Policy. Some of the major changes are: • providing guidance for the use of discretion when considering imposing daily civil penalties; • clarifying that a violation identified at any NRC-licensed facility with an approved Corrective Action Program may be closed-out as a non-cited violation (a violation for which there is no formal enforcement action) when certain conditions are satisfied; • adding a new section on civil penalties to individuals who release safeguards information; and • providing guidance regarding the notification of employers when the NRC discovers damaging or disqualifying information about an individual's trustworthiness and reliability. The revised NRC's Enforcement Policy, effective January 28, 2013, can be found here. Changes to the NRC Enforcement Policy since it was first published, with links to a summary of each change and the Federal Register notice for each change, are maintained on the NRC Office of Enforcement webpage. Questions regarding the Enforcement Policy revision can be directed to Lauren Casey at (301) 415-1038.

Comments

comment #53740 posted on 2013-02-08 00:39:43 by in response to comment #53637

Back and forth with generalities and distractions from the point forever. Malfunction is too slight a word, in describing the dangers, but I was trying to be brief. Earthquakes, for instance, are not predictable to any measure of damage expectations. If an earthquake strikes an Nuclear reactor site, shutting it down in time will be of no consequence. Undetectable amounts of radiation released after earthquake would be massive and devastating in so many ways. Not only will the reactors exposed components be releasing radiation toxins into the air water and soil, but the years of accumulated tonnage of spent Nuclear rods, being stored in the surrounding areas of the Reactor site, will be oozing toxins into the environment. I used the word undectable because nuclear radiation can not be detected by our human senses. You can not see it. You can not smell it. You can not taste it. It can only be measured by mechanical devices. When a mechanical device does measure radiation, it can only measured in an exact location. Pure theory has to be used to determine where if any, other radiation has traveled to. In South Carolina, water from rivers and resevoirs are cycled around the Nuclear reactors to cool them for operation. I say cycle because after the water is used to cool the reactors is retuned to the rivers. These rivers feed the water drinking supply down stream from the reactors. The NRC requires the Reactor site to check the water for contaminants and temperature before releasing it back to the river. I would have to assume the mechanical testing device would need to have the water pass over or through something to detect problems. I do not know this, but I believe the testing will not contain the water in small or large holding tanks and test each batch of water before releasing a batch. This batching method would be minimumly, although very costly and time consuming, the best way to do it. Maybe this can be confirmed by our NRC rep that reads this reply. If batching the water is not used, and the passing the water over a device is used, and when the water that is detected as unacceptable, there will be no control over how much water has already gotten out and back into the river. The recipient of the water coming from their tap won't see it, won't smell it, and won't taste it. Recently I read in the news in Florida, they are shutting down an aging Nulclear plant and it will take 60 years to shut it down. This shutdown is under a controlable arrangement. Sixty years ! Why ?! Because the toxins are so great that there is no way to get it done any sooner, apparently. Solar, wind, and tidal generated power can also have major problems, but never cause an irreversable affect to the human health and environment that Nuclear generated energy can. For me, I'd rather know when I am being poisoned by some toxic contaminant . I'll take my energy from renewable resources any day over Nuclear energy.

comment #56416 posted on 2013-02-15 18:33:39 by ralph

You have just been appointed as the director of a new overall health treatment or human provider firm (for example, a community well being clinic). As the director, a person of your first responsibilities is to draft a mission assertion and a values statement for your corporation. As your group grows, your stakeholders will deliver their input and assist condition these statements.

comment #53637 posted on 2013-02-07 14:59:04 by Anonymous in response to comment #53118

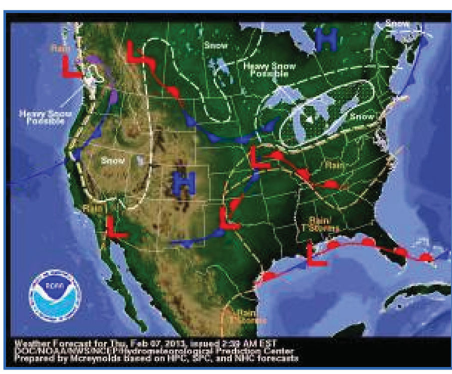
The consequences are underwhelming when undetectable amounts of nuclear contaminants escape capture. The increased risk to an individual from radiation released from a nuclear power plant is so small it is difficult to determine. It is far less than the risk incurred by an individual in riding in a car across a state for business or pleasure travel. Living exposes us to constant danger. A society is exposed to "constant danger" from using electricity. All forms of generation pose some risk. The pollution from generating electricity with fossil fuels causes health impacts including deaths on an ongoing basis. While nuclear power has its risks, I will take nuclear power's low probability risks any day over the real consequences from other sources of power.

comment #53118 posted on 2013-02-06 10:46:13 by

REGARDLESS OF NRC'S ATTEMPT TO ENFORCE ANY REGULATIONS, WHEN A FACILITY HAS BEEN ESTABLISHED, AND IS OPERATING, ANY MALFUNCTION WILL HAVE SOME IRREVERSABLE DAMAGE TO THE ENVIRONMENT. THE CONSEQUENCES ARE OVERWHELMING WHEN UNDETECABLE AMOUNTS NUCLEAR CONTAMINENTS HAVE ESCAPED CAPTURE AND HAVE ALREADY BEEN PASSED TO HUMAN INGESTION BY WAY OF DRINKING WATER AND FOOD FROM PLANTS IN CONTIMATED SOIL. THIS IS A CONSTANT DANGER WE HAVE TO LIVE WITH USING NUCLEAR POWER.

New England's Nuclear Power Plants Readying for Nemo

posted on Thu, 07 Feb 2013 20:03:05 +0000



Neil Sheehan
Public Affairs Officer
Region I

New England states and other parts of the Northeast are battening down the hatches in anticipation of a winter storm dubbed "Nemo" by the Weather Channel. Unlike "Finding Nemo," the 2003 hit movie from Disney featuring a clown fish dad roaming the seas in search of his wayward son, those in the storm's path won't have to look far to see its impacts. Indeed, forecasts are calling for blizzard conditions and upwards of two feet of snow in the Boston area. As with other significant storms, nuclear power plants that could be affected will be required to make preparations. These are actions such as ensuring that fuel oil tanks are adequately filled; that there are no materials on plant grounds that could become airborne missiles amid high winds; and that water-tight doors and other openings are properly closed in the event flooding becomes an issue. NRC inspectors stationed at all operating plants on a full-time basis will likewise be busy, as they independently verify the facilities – particularly the Pilgrim plant in Massachusetts and the Seabrook plant in New Hampshire -- are positioned for whatever wicked weather comes their way. To help guide those evaluations, the inspectors will follow a procedure and checklist focused on adverse weather protection. Once the storm arrives, plant operators have plans that guide their responses. For instance, if sustained wind speeds exceed a certain level, a plant would have to shut down. Also, if flooding were to be greater than pre-determined thresholds, an emergency declaration would have to be made and a shutdown may be necessary. During Superstorm Sandy last October, three nuclear power plants ended up shutting down for reasons that included high water intake levels and electrical grid disturbances, but all did so safely and effectively. As always, the work that takes place before the storm arrives is an essential part of ensuring any storm-related problems can be handled in a prompt, safe manner.

Comments

comment #54873 posted on 2013-02-11 12:25:36 by Moderator

> Thank you for your comments. We posted a press release on Saturday on PIlgrim's status. It's available here: http://www.nrc.gov/reading-rm/doc-collections/news/2013/#feb . Also, we just posted an update on this blog. Neil Sheehan

comment #54388 posted on 2013-02-09 16:07:43 by keithsbrooks@gmail.com

> Although it seems that the worst has been averted, I'd just like to insist that the NRC keep the public up to date as to the status of the Pilgrim Plant. I'm sure there's nothing to worry about, but I can't help but lose my resolve a tad bit when our chief nuclear regulatory agency decides to engage in "network testing" the day after a major storm (and consequently a potential nuclear dilemma). The nutjobs are already claiming that Nemo is a palindrome (or anagram — whatever it's called — I don't follow these theories), so I can't wait for you to update us on the status of the plant and quiet the wackos so we can all return to normalcy and give them a good old fashioned "I told you so". As far as federally tax-funded agencies go, your organization has always done an outstanding job at relating your findings to the public, and I'm sure that this will continue into the foreseeable future. Hopefully you've all weathered the storm safely — G-d willing of course. Thanks and enjoy what's left of the weekend, Keith

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Pilgrim in Cold Shutdown Due to Nemo-the-Nor'easter

posted on Mon, 11 Feb 2013 17:23:41 +0000

Neil Sheehan
Public Affairs Officer
Region I

True to forecasts, New England states bore the brunt of the winter storm dubbed Nemo. With respect to nuclear power plants in the region, only one – Pilgrim, in Massachusetts – had its operations interrupted by the powerful Nor'easter.  At 9:17 p.m. Friday, three off-site power lines that provide electricity for plant safety systems were knocked out of service. In response, the reactor, as designed, automatically shut down and the facility's emergency diesel generators activated to provide that power. One of the criteria for a plant to declare an "Unusual Event" – the lowest of four levels of emergency classification – is the loss of off-site power for more than 15 minutes. As such, Pilgrim made that declaration at 10 p.m. Friday. The NRC issued a press release early Saturday morning. After one of the lines was restored, the plant was able to terminate the Unusual Event as of 10:55 a.m. Sunday. But there was a setback later in the day when the 345-kilovolt line experienced new problems. Once again, the emergency diesel generators started and will supply the power needed for safety systems until the lines are fully restored. Since the reactor was already in "cold" shutdown condition, Pilgrim did not need to again declare an Unusual Event. NRC inspectors, and for a good part of the weekend the NRC Region I Incident Response Center, closely monitored the storm recovery efforts at Pilgrim. That will continue as repair work is carried out and plans for placing the unit back in service are developed.

Comments

comment #56372 posted on 2013-02-15 13:56:32 by Moderator in response to comment #56339

> The ground elevation of the Protected Area buildings at the Pilgrim nuclear power plant is 23 feet above sea level. The plant's emergency diesel generators (EDGs) are housed in a structure that provides robust protection from the elements and is designed to withstand flooding, seismic activity, missile hazards and more. Also, a breakwater exists along the facility's canal and intake structure that serves to lessen the impact of storm wave action on the intake structure and shoreline revetment. Flooding studies done for the Pilgrim site evaluated the impact of postulated significant storm surge events. They concluded that no flooding would occur that could affect buildings containing nuclear safety-related equipment, including the EDGs. We are not aware of any storm surges that have impacted any Protected Area buildings at Pilgrim since it came online in 1972. We would point out that as part of the NRC's multiple post-Fukushima actions, the NRC is continuing to reassess the ability of U.S. nuclear power plants to withstand severe flooding. Visual inspections of all critical infracture, including the EDG building, are part of this process. The results of those reviews will be made available to the public once they are completed. Neil Sheehan

comment #56339 posted on 2013-02-15 11:39:24 by Norm

> Is it true that the plant and its emergency generators are only about 20 feet above sea level and that now all it would take is for a storm with a good size swell to take them out, creating the fukushima style problem?

comment #56409 posted on 2013-02-15 17:34:16 by Fred Stender in response to comment #56372

> What about the fuel storage tanks, I don't mean the day tanks, the bulk tanks. At Fukushima, the tanks were placed right on waters edge, for ease of refueling from ship. They were scrubbed off the second the tsunami hit. An underground storage tank always has

vents, so underground isn't the solution either.

comment #55500 posted on 2013-02-13 10:40:51 by in response to comment #55195

Mr. Greenidge, I guess you believe there is no other alternative for energy than Nuclear. Well, surprise, there is. This country is sadly behind in the technologies of solar wind and other renewable resources. It's easy to go along with what you can't change. Oh yes, oh yes, breathing coal and oil toxins in the air is bad, no doubt. Conversely, you would never know you are ingesting, or breathing Nuclear toxins. You can't see it. You can't smell it. You can't taste it. It's only detectable by a mechanic device. That, of course, appears to be ok with you. It is not ok with me. We do have other Natural resources available for creating energy. The simple fact is that if the Nuclear industry does not figure a way to re-use nuclear waste, we will absolutely run out of space to put it. And, like you said, you can volunteer your backyard for it. James Matthews Columbia SC

comment #55535 posted on 2013-02-13 13:54:16 by nuclear guy in response to comment #55195

Mr. Mathews, Nuclear power plants have devices which record any radiation release from the facility during normal and abnormal opeartions. All releases are recored and submitted to appropriate agencies including the EPA and NRC on an annual basis. Levels exceeding regulatory limits are reported and require actiosn to be taken up to and including shutdown of the facility in order to terminate the release. Plant buildings are kept at a vacuum with respect to the atmosphere so that radioactive material is held within the building to be filtered and prevent or limit released to the environment such that only heavily filtered releases occur, and levels are limited in nature. The combination of these methods, monitoring, a bias for action, filtering, all prevent uncontrolled releases of radioactivity and assure adequate protection of public health and safety. Your concern about radiation is a valid concern, however one needs to go well beyond nuclear plants to see radiation levels that could harm public health. The average person receives about 320 mRem of radiation per year. A person living near a nuclear power plant typically receives less than 1 mRem of additional radiation. In the grand scheme of things this is negligible to public health. For comparison, flying across country will give you greater than 1 mRem of exposure, as will X-rays, CT scans, cardiac cath/Tc-99 and thyroid iodine treatments, spending a lot of time in concrete/granite based buildings, or living by a coal plant. All of these are things the public does on a daily basis and considers a negligible risk. I encourage you to check out the radiation facts page on the nrc website for more details.

comment #55541 posted on 2013-02-13 14:52:04 by Fred Stender in response to comment #55195

Mountain out of a mole hill I just discovered a new type of false argumentative technique, its called "Trivializing the Issue" Even in a geologically stable deep storage situation, the chance of something going awfully wrong is still present, and the results could be extinction level, so no, this is not trivial.

comment #54883 posted on 2013-02-11 13:19:39 by nuclear guy in response to comment #54875

Per their operating license, they are requried to have over 30k gallons of fuel on hand per generator. This is over 1 week per generator at full load. The standard for nuclear plants is a requirement for 7 days of fuel per generator, unless it takes longer than 7 days to get resupplied fuel to the plant. Then it takes 7 days + extra resupply time.

comment #54875 posted on 2013-02-11 12:38:40 by Fred Stender

How many days of diesel do they have on site right now, and what will that power? Please be specific under various scenarios, thanks!!

comment #54907 posted on 2013-02-11 15:21:00 by Moderator in response to comment #54875

The Pilgrim plant has sufficient fuel oil to operate its emergency diesel generators for more than a week. The tanks used to store that fuel can, of course, be refilled as needed. Efforts to restore off-site power to the facility are ongoing, with the objective of returning to that normal mode of power supply as soon as possible. In the meantime, the emergency diesel generators can meet all of the power needs for key safety systems. Neil Sheehan

comment #55982 posted on 2013-02-14 10:30:38 by in response to comment #55535

Mr. Nuclear guy, There are tons of words to be spoken for and against Nuclear energy production. And, as you say, having factual information is pretty important. I do believe there are all sorts of measuring devices to detect levels of radiation working to prevent any problems. The problem is these devices were created by humans and are monitored by humans. The cherry on top is the whole process is run by a government designed arrangement. I worked for my state government for 11 years and was in the was in the military for four years. I have a pretty good idea of how things work with the government and it does not make me feel better about Nuclear energy. Of course, any monitoring is better than none but to question a government process is fruitless. Any change or attempt to investigate a concern from lowly citizen is satisfied with a few bread crumbs and a whole bunch of words. I know I have attended and spoke at several Public Service Commission meetings. I have an engineer friend that works for a crane company that installed the cranes at many of the Nuclear reactor sites. These cranes are used to remove and insert the rods that run the reactors. There was a time he had to inspect one of the cranes that was malfunctioning. He was required to attend 1-2 weeks class about the environment before he could check out the problem. Of course, this is a good thing. It also indicates how dangerous the environment is. This fact and what great detail and special handling is required to store spent Nuclear rods is all I need to know about Nuclear energy that it is a bad idea. You spent time doing the radiation comes form everywhere speel. This, as always, does not take away, but also adds to my fears, that Nuclear energy production is bad for our environment. I know you feel the need to talk up your field of

Nuclear development. Everyone needs to justify their job. All I know is there are so many natural energy resources "above ground" that are in endless supply and all we need to do is harness it. Oh yes, oh yes, there can be accidents with solar and wind harnessing devices, but these accidents will not cause an endless destructive domino affect on the environment.

comment #57929 posted on 2013-02-19 14:57:00 by Moderator in response to comment #56372

The Pilgrim emergency diesel generator (EDG) fuel storage tanks are buried underground, just below the 23-foot above-sea level elevation, underneath an asphalt overlay similar to a parking lot. These tanks are vented through piping that runs vertically along the height of the EDG enclosure building for approximately 12 feet above the 23-foot above-sea level site elevation. As such, there would have to be an event resulting in water intrusion at the site at least 35 feet above sea level for the tank vents to be affected. During a response to comments left on the blog post last week, we pointed out that there had not been an event in the history of the Pilgrim plant that led to flooding at the 23-foot above-sea level site elevation. The plant came online in 1972. Neil Sheehan

comment #54921 posted on 2013-02-11 16:34:02 by Fred Stender in response to comment #54907

Thanks Neil

comment #55195 posted on 2013-02-12 18:17:08 by James Greenidge in response to comment #55111

Nuclear waste is a political issue and a technical non-issue. Any high school grad can tell you how to put truly exhausted fuel away. It's wind-vane politicians and a skittish science illiterate public making a mountain out of mole hill on putting it in places it ought to go. Till then, I'd rather babysit several thousand tons of quiet nuclear waste that isn't going anywhere than happily breathe in tens of millions of tons of oil and coal waste that greets you inside your own homes every day. James Greenidge Queens NY

comment #55111 posted on 2013-02-12 10:33:46 by

I was referred to this site anonymously from the NRC after I sent the NRC a letter to thme about my concerns with Nuclear waste. I am personally against Nuclear energy because there is no way known yet to dispose the tonnage of Nuclear waste produced from the electrical energy. From this posting, all I can say is yes, oh yes, you all did a good job, as in this is your job and it is required. I know having a job, any job, today, is very important. The problem is, I can not speak to anyone that can make any difference about discontinuing Nulear energy. Of course, it is too late, now that we have all this safety stuff in place. Where was I when all the decisions were being made about going ahead with developing Nuclear energy ? I was working my ass off trying to raise a family, like everyone else in the same position. We trusted the powers to be to make the right descions for us because we did not have the power to stop it. Now we are stuck with Nuclear power and I hate the fact that I can't do anything about it. So you dam well better do a good job because this is what the powers to be decided for us !!!

NRC Sustainability Plan Shows an Agency Committed to the Environment

posted on Wed, 13 Feb 2013 15:57:06 +0000

Ian Fisher
Sustainability Manager

As required by Executive Order, the NRC submits an annual Sustainability Plan to the White House and OMB. This plan outlines the agency's plans to be a good environmental steward through efforts to reduce greenhouse gases, increase use of renewable energy and reduce gas consumption. The NRC has posted its plan online. It reflects the agency's commitment to conducting our agency business in an environmentally and responsible and sustainable way. In short, the NRC met or exceeded all relevant local, state and federal environmental laws and regulations, continually enhanced our business practices to minimize environmental impact, and sought to manage the agency's ecological footprint by reducing the use of natural resources and preventing pollution. As the plan shows, during FY2011, the agency implemented and continued a number of energy-reduction projects. We made changes to the air conditioning system at our Rockville, Md., headquarters that allowed us to cool the buildings more efficiently. We expanded our telework program which, in turn, reduces the number of cars commuting to work. The agency also is upgrading its restrooms with water saving toilets and faucets, which may save as much as 10 to 15 percent of water usage. An additional office building in the White Flint headquarters complex was also built with aggressive energy efficiency and "green" technologies in mind. We hope you'll take a minute to review our latest update.

Comments

comment #56350 posted on 2013-02-15 12:31:12 by american2018@gmail.com

You are upgrading toilets to save water, toward sustainability, while your industry has released a thousand tons of nuclear fuel into the northern hemisphere, and you are breathing MOX. Why don't you put your engineering skills and money to use in better things, like a) finding a microbe that can do what Brown's Gas flame does to radioactivity, and b) making photovoltaics work in the dark via cosmic rays, make solar pv's more efficient and cheep, break out the technology now, make wind generation economical, and solve or put into deployment the technologies to store electricity efficiently.

comment #56361 posted on 2013-02-15 13:12:13 by Moderator in response to comment #56352

All comments that adhere to the blog comment guidelines are approved and posted within 24 hours during the work week, usually sooner. Moderator

comment #56352 posted on 2013-02-15 12:32:40 by american2018

"Your comment is awaiting moderation", I thinks = "Your comment is awaiting censorship."

comment #56353 posted on 2013-02-15 12:33:26 by american2018

HELL FIRE you're worried about sustainability while awaiting a Carrington Class solar flare and a New Madrid Valley 15 reactor pipe-breakage death of the nation?

comment #56351 posted on 2013-02-15 12:32:00 by american2018

You're upgrading toilets toward 'sustainability' while breathing MOX particulates from Japan?

Licensing Project Managers – The NRC's Expert Generalists

posted on Fri, 15 Feb 2013 13:13:43 +0000

[caption id="attachment_3801" align="alignright" width="225"] Lauren Gibson[/caption]
Lauren Gibson
Licensing Project Manager

A licensing project manager for an operating reactor has a lot of responsibilities. We coordinate technical reviews, interface with the licensee, support the regional staff and the resident inspectors, respond during any incidents, and serve as the headquarters point of contact for anything related to the plant. We're basically expert generalists who need a solid understanding of all the operational and licensing situations for each plant– and, importantly – the ability to communicate about it all. I've been a licensing project manager for about three years, currently for the Palo Verde nuclear facility in Arizona and Columbia Generating Station in Washington State. The majority of my time is spent coordinating the review of license amendment requests. A plant's license is much more complicated than your driver's license. It does far more than just grant permission to operate a nuclear facility. The license also specifies how it is to be run. Imagine having a driver's license that required you to keep the gas tank more than half full and all four tires inflated to a certain pressure. If you wanted to wait until only a quarter of a tank remained to get more gas or you wanted to change the pressure level in your tires, you'd have to apply to amend your license. I am the person who receives those equivalent requests for my particular sites. Those requests are then reviewed by a team of appropriate scientists, engineers, and experts, with the project manager responsible for engaging those experts and keeping an eye on the entire review. The project manager is in charge of coordinating schedules, interfacing with the licensee (the operator of the plant), ensuring that public documents are written in plain English, and packaging the final approval or denial of the request with a clear justification. In my experience, licensees have four to12 such requests at the NRC at a time. The project manager needs to understand the technical aspects of each one. The project manager also needs to be aware of conditions at the site. The project manager serves as the headquarters point of contact for all matters related to the site, so it's very important to know what's going on. Every workday, I participate in a conference call with the resident inspectors and the region to discuss plant status and concerns. If the region or the residents need any headquarters support, I am there to provide it directly or to arrange it. One of the most important and, thankfully, infrequent duties of the project manager is to respond in case of an incident at the site. If the headquarters operations center is activated, I will report there and provide site specific information. Having someone there who is familiar with the site, the conditions at the site, and any licensing and operational issues is important. We also have the pleasure and responsibility of ensuring that the NRC is being open and transparent to the public. Licensing project managers run many of the site-specific headquarters public meetings. So, while licensing project managers need to understand the highly technical aspects of a review or a performance issue, we also need to see the big picture, and how one issue may relate to other issues or actions at the plant. And we have to communicate it to all stakeholders – both inside and outside the NRC. It's a great job, and one I'm happy to be doing. But it's not easy being an expert generalist.

Comments

comment #56338 posted on 2013-02-15 11:37:24 by kurtpenner

> Lauren - your blog precipitated a thought and concern as to the effectiveness of resident inspectors at regulated facilities and as to their responsibilities to prevent or curb a licensee from underperforming and violating licensee requirements. Are the NRC's routine inspection findings and performance indicators capable of identifying repetitive degradations of single cornerstones as per your ROP matrix? For example, should the NRC have identified the numerous performance and technical issue findings of Fort Calhoun Station in Nebraska BEFORE the Missouri River flooding and eventual electrical switchgear fire?

comment #56404 posted on 2013-02-15 16:51:24 by arthur rosenzweig Yale 53

> so what happened with Pilgrim nuclear, where you and they now stand accused of criminal ..colluson to hide the inability of Cape Cod --an island--to evacuate.

comment #56390 posted on 2013-02-15 14:50:34 by Moderator in response to comment #56276

> In response to the decision in June 2012 by the U.S. Court of Appeals for the DC Circuit, the Commission decided to stop all licensing activities that rely on the Waste Confidence Decision and Rule. The NRC created a Waste Confidence Directorate to oversee the drafting of a new Waste Confidence Environmental Impact Statement and Rule. The Commission has instructed the Directorate to issue the final Environmental Impact Statement and Rule by no later than September 2014. For now, the staff have been instructed to continue their reviews, but no new licenses will be approved. Here is a link to the project impacted: http://pbadupws.nrc.gov/docs/ML1227/ML12276A038.pdf For an explanation of the waste confidence decision and rule, please see the blog post "Deciphering the Waste Confidence Order" from August 9, 2012 (http://public-blog.nrc-gateway.gov/2012/08/09/deciphering-the-waste-confidence-order/). The licenses for the new Vogtle and Summer reactors are valid, as is the construction permit for the unfinished Watts Bar 2 reactor, so construction at those sites is acceptable. Lauren Gibson

comment #56392 posted on 2013-02-15 15:06:57 by kurtpenner in response to comment #56387

> Lauren - I appreciate your reply, but my question is very specific and one of your colleges might be better suited to answer. Why did it take an unusual event to occur before the NRC identifies numerous performance deficiencies (techincal issues, safety attitude, performance attitude, etc.)? What are the responsibilities of a resident inspector? Thank you!

comment #56388 posted on 2013-02-15 14:49:17 by Moderator in response to comment #56315

> The NRC communicates with stakeholders as part of its commitment to conducting its regulatory process in an open and transparent way. This includes communicating with the public, many of whom are ratepayers. More information on our Open Government initiatives is available here: http://www.nrc.gov/public-involve/open.html Lauren Gibson

comment #56387 posted on 2013-02-15 14:48:35 by Moderator in response to comment #56338

> In addition to the resident inspectors at each plant, special inspections of various types may also be initiated in response to possible deficiencies at plants. The Reactor Oversight Process is outlined here: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html . It offers stringent inspections based on plant performance. Since I am not the manager for Fort Calhoun and can't speak directly to the technical findings at that plant. For more information on oversight at Fort Calhoun, please go here: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/FCS/fcs_chart.html Lauren Gibson

comment #56450 posted on 2013-02-15 23:23:11 by in response to comment #56390

> It is still unclear why an incomplete Nuclear site has a license to continue building when waste confidence is based on exsisting structural ability and space to store waste. Is it being based on proposed space alone and not on the facility to store the waste ? I can not see anyway it can be determined that there will be enough space to store Nuclear waste, regardless, especially in areas in the north east part of the country where populations are so densely populated. I would think the tonnage of waste that is created yearly in the north east is already encrouching unsafe distances from residential areas. This is the point. It is totally irresponsible to believe that there will be enough space to store nuclear waste indefinitely. I don't care what diety decided it was ok to continue building sites ! You tell those pompous judges to have that nuclear waste put in their backyard and see how that flys !This is the reason to stop building more Nuclear sites. James C. Matthews Columbia, SC

comment #56315 posted on 2013-02-15 11:14:26 by Diane Smith

"And we have to communicate it to all stakeholders – both inside and outside the NRC." Why? The stakeholders don't seem to be paying for anything anymore. The costs of nuclear plants are being sloughed on ratepayers. We're living in the age of "corporate welfare".

comment #56279 posted on 2013-02-15 10:20:34 by Guillermo Alcocer

I see that's a really complex job. Thanks for share your experiences with us.

comment #56276 posted on 2013-02-15 10:13:17 by

Ms. Lauren Gibson (Licensing Project Manager) I read what your duties are and maybe you can give me a definitive answer. I understand there is a freeze on any new licensing for building New Nuclear reactor sites in the USA. I would like to know exactly why, in as few words as possible, why this freeze has occurred. In addition, and in conjunction with the freeze, I am confused as to why any incomplete Nuclear reactor site building is still allowed to continue. If there is a physical reason, such as lack of storage of nuclear waste due to the shutdown of the Yucca Mountain storage faculty for nuclear waste, then why would construction of new reactor sites continue ? I hope the reason is not because the incomplete reactor sites already have their license. James C. Matthews Columbia, SC

comment #58240 posted on 2013-02-20 11:28:42 by Moderator in response to comment #56338

The NRC inspection program is proactive: it is meant to identify and correct problems before they become a significant safety concern. At Fort Calhoun Station, a regulatory hold was put in place effective Sept. 2, 2011, because the NRC saw the need for increased oversight. Increased oversight, known as the IMC 0350 process, will provide reasonable assurance of public health/safety prior to plant restart. While it's true that Fort Calhoun has had performance issues, the NRC is engaging the plant operators to address these concerns. In fact, it was the NRC that identified a flood mitigation issue back in 2009 prior to the record floods in 2011. We are confident in our oversight process and will continue the important work of ensuring all issues are addressed by the licensee. Lara Uselding

comment #59044 posted on 2013-02-22 08:56:48 by in response to comment #58240

OH YES, OH YES, NRC CAN AND DOES DO ALL IT CAN TO PROTECT THE HUMANS. THEY'RE SUPPOSED TO. AND, IF WHAT THEY DO, ISN'T ENOUGH, WELL WE'RE ...... I'D LIKE TO KNOW WHAT NRC WILL DO WHEN ALL THE ALLOCATED SPACE FOR THE NUCLEAR WASTE HAS BEEN USED UP. PASS THIS ON TO ANYONE WHO CAN MAKE A DIFFERENCE. Jim Matthews, Columbia, SC

comment #57290 posted on 2013-02-18 09:12:21 by Kurt Penner in response to comment #56387

I appreciate your reply, but my question is very specific and one of your colleges might be better suited to answer. Why did it take an unusual event to occur before the NRC finally identified numerous performance deficiencies (technical, safety attitude, performance attitude, etc.)? What are the responsibilities of a resident inspector? Thank you

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The Online Public Meeting Schedule – A Resource for the Interested Public

posted on Tue, 19 Feb 2013 18:02:46 +0000

Adam Glazer
Librarian, Public Document Room

The NRC hosts hundreds of meetings throughout the year. Many of the meetings are held so you, the public, can share your thoughts about nuclear power issues. While the meeting topics vary, the way to find out about them doesn't -- you check the Public Meeting Schedule on the NRC's Web site. You'll find the date and time, purpose and agenda, location, and contact name and phone number. When you click on the link to the agenda, you'll be able to find out more information, such as who from the NRC is planning on attending the meeting. If there's a telephone icon, there will be a phone number so you can listen in on the meeting remotely instead of traveling to it. Our goal is to give you at least 10 days advance notice before a meeting, so that you can arrange your schedule to participate if it's a topic you're interested in. A word of caution – please keep checking the Web site in case there's been a change to the meeting. Also, if there's bad weather, we may have to cancel or postpone the meeting. In 2012, we posted information about 1,147 meetings. There are sure to be many meetings in 2013. Perhaps one will interest you?

Comments

comment #57917 posted on 2013-02-19 14:31:05 by Sardar

On this site, we post new meeting notices and we also keep information up to date by updating changes to upcoming meetings.

comment #57896 posted on 2013-02-19 13:08:26 by Steven Dolley

Some NRC meetings are closed to the public, to protect security-related or proprietary information, among other reasons. When a meeting must be closed, NRC staff should strive to release a meeting notice well in advance of the meeting, even though it's not open, because the fact that such a meeting is held is of interest in itself. Frequently NRC does not release notice of closed meetings until after they take place, if ever, which is not in the interest of transparency.

comment #58229 posted on 2013-02-20 10:50:37 by

I clicked on the link from your email to me to see the Public meetings and it opened the NRC site and stated Page not found. 2/20 10:50am est.

comment #57938 posted on 2013-02-19 15:21:55 by CaptD in response to comment #57896

Good suggestion, they should also post a Non-Secret summation of all closed meetings as many from the public view these closed meeting as a dodge to public oversight!

comment #57941 posted on 2013-02-19 15:26:55 by CaptD

Every post should include BOTH the webinar LINK and also the Telephone Only Link to the meeting , not to yet another listing, along with who to contact if the connection is poor during the meeting and/or how to submit questions during the meeting! Another suggestion might be to include a link to be noticed when the video and/or transcript are posted for any meetings of interest.

comment #58237 posted on 2013-02-20 11:15:43 by Moderator in response to comment #58229

The links in the post are good (we just checked), but here is the link to the public meeting page: http://www.nrc.gov/public-involve/public-meetings/index.cfm Moderator

comment #58307 posted on 2013-02-20 14:33:28 by Moderator

These are excellent suggestions. The NRC is currently changing how staff posts public meeting information. Although we may not be able to take action on comments such as these before the new process goes into effect, we could very well make further changes to the system and process at a later date. Lance Rakovan Facilitator Program Coordinator

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How the NRC is Responding to the Cooling Water Leak At the Palisades Nuclear Plant

posted on Wed, 20 Feb 2013 16:12:02 +0000

Prema Chandrathil
Region III Public Affairs Officer

On Friday, the Palisades plant in Covert, Mich., shut down so plant personnel could find and repair a leak somewhere in the reactor's cooling water system. Soon after, the NRC dispatched an additional inspector from the Regional III office, located in Lisle, Ill., with a background in mechanical testing and repairs. He supplements the two NRC resident inspectors as they evaluate the plant's repair activities.

For more than a week now, the NRC resident inspectors on site have been following the actions taken by workers at Palisades to find the leak. The resident inspectors reviewed the data. They also watched plant workers as they isolated different parts of the system to conduct tests to try and identify exactly where the leak was coming from. Plant workers caught the problem because the water level in the component cooling water system was going down slowly. This system uses non-reactor water to cool certain safety equipment. Per NRC regulations the system is required to be monitored. When the plant shut down the system was leaking about 35 gallons per hour. This water was captured and released to Lake Michigan through an established monitored release path. The leak did not place the plant or the public in danger. It's now believed a heat exchanger in the system is the source of the leak. A heat exchanger is basically a box that contains around 2,000 tubes. The tubes have water running through them to remove heat from equipment, such as seals or pumps. This heat exchanger plays an important role to cool necessary equipment during normal operation, but also during potential accident scenarios. Palisades has two safety-related heat exchangers in this particular system; both are required by NRC regulations to be in working condition and ready to respond at a moment's notice. With one of the two exchangers potentially not working right the plant decided to shut down before the regulations required it. NRC regulations state if there is a problem with the heat exchanger it would need to be fixed within 72 hours. If that's not possible the regulations require the plant to shut down to find the leak and make the appropriate repairs. The plant will only be able to restart when the heat exchanger is working correctly. Over the weekend all three NRC inspectors continued to monitor and assess the repair work to find and fix the leak. The NRC will continue to closely follow this event and observe how the plant goes about these activities with safety in mind from start to finish. We know the community is interested and concerned about these types of issues and continue to work to keep our commitment to ensure they are informed. One of our initiatives is to provide summaries of conversations between the NRC and plant staff to the public. A summary of such a conversation about this leak, which took place on Thursday, Feb.14, will be available to the public in the near future. Our assessment of this issue will also be documented in a publically available inspection report.

Comments

comment #58504 posted on 2013-02-21 01:51:19 by joffan7 in response to comment #58260

> Absolutely correct Jeff. Of course the same logic should also be applied to San Onofre; now that the steam generators have been inspected and all worn tubes plugged, those reactors should also be up and running again. That won't happen either. The NRC appears to have no interest in total societal risk, only in the subset of risk attached to nuclear plants without regard to the countervailing benefit.

comment #64588 posted on 2013-03-01 13:10:50 by Moderator in response to comment #64270

> Additional information can be found in the NRC's preliminary notification and in the meeting summary of the call our staff had with the plant to discuss the details regarding this issue. The NRC does not normally summarize these types of conversations but we know the community is interested and concerned, and we want to be as open and transparent about these types of issues. Here are both numbers to search in the agency's public document system ML13052A640 for the meeting summary and ML13053A365 for the preliminary notification we issued. Once the inspection report is complete it will also be made publically available. Prema Chandrathil

comment #64270 posted on 2013-02-28 17:45:19 by David Greene C.Phys; F.Inst.P; F.Inst. I.&C.

> I would like to get more detailed information on the heat exghanger leak incidents. I have a background in NPP fluid dynamics and heat transfer, as well as NPP I&C. Can you provide further information and/or references to technical issues and/or technical personnel associated with this incident.

comment #58260 posted on 2013-02-20 12:31:56 by Jeff Walther

> "with safety in mind from start to finish" It seems to me that in the big picture it would be a lot safer to the people in the area and the environment, if the plant continued to operate, even with a small harmless leak like this, rather than replacing it's clean electricity generation with the pollution from fossil fuels. Shutting down the plant may be extra safe from the plant's point of view, but in the big picture, shutting down the plant for trivial problems is incredibly damaging to the environment, when the replacement electricity generation will spew particulates and CO2 into the environment. Get it operating again as quickly as possible.

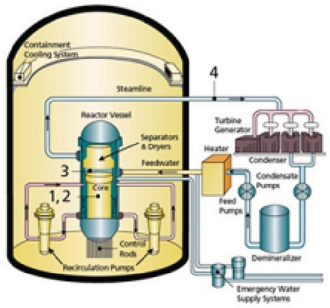
Thermal Hydraulics: Heat, Water, Nuclear Power and Safety

posted on Fri, 22 Feb 2013 15:38:27 +0000

Scott Krepel
Reactor System Engineer

One of the most important safety questions in a nuclear power plant is: Can you cool the very hot nuclear fuel in an accident when normal cooling is disrupted? The scientific field best equipped to answer this question is called "thermal hydraulics."

Typical Boiling-Water Reactor



Source: U.S. Nuclear Regulatory Commission

The first part of the term, “thermal,” relates to heat transfer, such as the movement of heat from the burner on a stove to the water in a pot via the metal of the pot. The second part, “hydraulic,” relates to the flow of a fluid such as water. The combination, “thermal hydraulics,” can be applied to systems where both the flow of fluid and the transfer of heat are important – such as a nuclear power plant. I work in the NRC’s Office of Nuclear Regulatory Research as part of a team dedicated to expanding our understanding of thermal hydraulics and applying that understanding in nuclear power plant safety. Over time, we’ve put much effort into incorporating existing knowledge into the NRC’s thermal hydraulics computer simulation program, TRACE. This program allows NRC staff to construct computer models of the cooling systems of a nuclear power plant and then simulate accidents such as pipe breaks (but not wildly improbable events such as the considerable destruction caused near the end of a typical superhero action movie). TRACE is constantly being pushed to become more accurate, reliable and versatile. Universities and test facilities around the world are conducting experiments and accident simulations to collect real-world data that can be used to determine TRACE’s ability to accurately predict specific phenomena. We use the outcomes to update the program as needed to make it more accurate and to better capture certain phenomena. Sometimes, new safety issues may result in further investigation of certain scenarios and further evolution of TRACE. Ultimately, the goal of this work within the research arm of the NRC is to continuously expand our understanding of situations which may impact the cooling of the nuclear fuel. This knowledge can then be used to ensure that the public and the environment are protected in the unlikely event of an accident at an U.S. nuclear power plant.

Comments

comment #63269 posted on 2013-02-26 22:57:13 by american2018@gmail.com

> Lots of talk. The reality is the atmosphere, the Pacific Ocean, and the dairy pastures on the west coast are all now contaminated. Here's a question: are the pumps at Fort Calhoun submersible, since they are on a flood plain? Here's another: What's the NRC done about the faked quake test results that whisteblowers reported? Here's another: How many babies died in the womb, or shortly after birth from Fukushima fallout?

comment #65970 posted on 2013-03-05 10:47:51 by Aladar Stolmar in response to comment #59567

> There is only one reference to zirconium in NRC-2012-0022-0004: “These pellets are stacked and sealed inside long, slender, zirconium metal-based alloy (Zircaloy) tubes to form fuel rods”. Zircaloy Mass in Fuel Cladding [kg / lb] 16,465/ 36,300 in the PWR and 40,580 /89,500 in BWR from NRC-2012-0022-0002 and NRC-2012-0022-0003. Zr (91) + 2 H_2O (36) = ZrO_2 (123) + 2 H_2 (4) + 5 MJ/kgZr Water required for complete reaction for the PWR 16,465 * 36/91 = 6513,6 kg or about 6.5 m3 (available), it produces 16,465 * 123/91 ZrO_2 = 22,255 kg zirconium dioxide and 16,465 * 4/91 = 723.7 kg Hydrogen and 82,325 MJ heat. For a 10 second firestorm duration it gives 8GW power… or twice the full power of the reactor… Water required for complete reaction for the BWR 40,580 * 36/91 = 16053,6 kg or about 16 m3 (available), it produces 40,580 * 123/91 ZrO_2 = 54,850 kg zirconium dioxide and 40,580 * 4/91 = 1784 kg Hydrogen and 204,250 MJ heat. For a 10 second firestorm duration it gives 20GW power… or five-six times the full power of the reactor… Considering that NRC does not require a top of the reactor depressurization vent to prevent the zirconium firestorm in the reactor, the above back of the envelope calculated worst case scenario should be considered.

comment #64818 posted on 2013-03-01 22:24:32 by The Firestorm Fighter in response to comment #59131

> Nothing that you say provides any evidence as to why you believe the NRC does not consider the effects of zirconium oxidation. Please point to a specific regulation that you believe should be changed and how you would change it. The fact is that the NRC DOES consider the effects of the "zirconium firestorm", as you so affectionately, but incorrectly, characterize it. Have you ever looked at the regulations in 10CFR 50.46? Paragraph 50.46(b)(1) requires that the calculated maximum temperature of fuel element cladding not be ~ greater than 2200'F. (A limit established so as to prevent both runaway clad oxidation and clad embrittlement). In addition, paragraphs 50.46(b) (2) through (b) (5), which contain required limits for calculated maximum cladding oxidation and maximum hydrogen generation, require that calculated changes in core geometry remain amenable to cooling and that long-term decay heat removal be provided. In addition, there is Regulatory Guide 1.157, which describes models, correlations, data, model evaluation procedures, and methods that are acceptable-to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a loss-of-coolant accident. It says the following on page 6: 3.2.5 Metal-Water Reaction Rate The rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam should be calculated in a best-estimate manner. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. For rods calculated to rupture their cladding during the loss-of-coolant accident, the

oxidation of the inside of the cladding should be calculated in a best estimate manner. 3.2.5.1 Model Evaluation Procedure for Metal-Water Reaction Rate. Correlations to be used to calculate metal-water reaction rates at less than or equal to 1900'F should: a. Be checked against a set of relevant data, and b. Recognize the effects of steam pressure, pre-oxidation of the cladding, deformation during oxidation, and internal oxidation from both steam and U02 fuel. The data of Reference 11 are considered acceptable for calculating the rates of energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900 'F.

comment #62440 posted on 2013-02-25 09:34:32 by David Andersen. in response to comment #59216

I think you may be a little confused about hoe a pressurized water reactor plant works. the dry steam pockets occur in the secondary side of the steam generators, The primary side where the coolant pumps are is pressurized and keeps the water in the system sub cooled. Steam pockets will not occur in the primary side of the S/G tubes under normal operating conditions.

comment #59131 posted on 2013-02-22 11:20:04 by Aladar Stolmar

Once the NRC allows the consideration of the real process, the zirconium firestorm in the core, reducing the steam and generating large amount of hydrogen, it will become pressing that the steam bubble covering of the fuel has to be avoided at any cost. Which is possible by venting and depressurizing the reactor and providing sufficient water reserves for gravity injection.

comment #65966 posted on 2013-03-05 10:36:17 by Aladar Stolmar

Nuclear guy and The Firestorm Fighter - the issue is that the real process when a stagnant steam bubble forms in the reactor core is an ignition and firestorm of the zirconium-steam reaction as it was modeled in the cited experiment as well as in the PBF SFD tests, most correctly in the scoping test of this later series. Which was indeed the real process in the TMI-2 accident, Chernobyl-4 accident, Paks 2 refueling vessel incident and in Fukushima 1, 2 and 3 reactors. What I want to achieve is the addition of depressurization vent and operator response to depressurize the reactor and avoid the steam bubble formation in the core of BWR and PWR types, therefore exclude the possibility of the firestorm. Until the NRC talks about core meltdown, steam explosion and who knows what other make-believe processes and use incorrect models, defining the cladding temperature incorrectly for the steam bubble covered state and until the NRC pushes for Hydrogen recombiners for the supressed Hydrogen generation rates calculated by incorrect codes, instead of looking into the real ignition and burning of Zircaloy cladding in the steam there will be no safe nuclear power plant. Even it would be relatively easy to achieve: the hot leg side injection ports could be used for the depressurization vent lines in the PWRs and the safety relief line in the BWRs. Whenever the state of the reactor is unknown, the forced circulation through the core is lost or the heat transfer to the ultimate heat sink is severed the operators should open the depressurization vent and allow the reactor to depressurize rapidly and the staged injections of borated water to operate. Only question remains: do we have sufficient gravity emergency core cooling water reserves? And yes, NRC is responsible for the Fukushima. "Even "Zirconium being one of the strongest reducing agents in the periodic table" page 148 of NRC-2012-0022-0002 "At the same time, Zr is also reacting with steam from concrete decomposition, producing hydrogen gas," Zr + 2H2O = ZrO2 +2H2 But the reaction heat of 5 MJ/kg Zr reacted is missing as well as the reaction of steam not from concrete, but the coolant itself! WHY? NRC does not allow the steam from coolant react with the zirconium, just with the steam from the concrete?! And the nature just follows the orders of NRC?! As we saw it in Fukushima Daiichi, indeed!" - cited one of my comments...

comment #65679 posted on 2013-03-04 11:01:35 by Aladar Stolmar in response to comment #64818

The Firestorm fighter does not recognize the difference between the best estimate model with two sided oxidation of solid cladding and the real firestorm ignited and burned in the reactor cores, zirconium reducing the steam, resulting in hydrogen and zirconium dioxide reaction products. Sad, but the NRC longstanding position is the tragic situation, which led directly to the Fukushima disaster. As what You are calling for an evidence... "TEST RESULTS Following the uncovering and dryout during the coolant boilaway, the rods heated at a rate of 2 to 5 K/s until peak cladding temperatures of 1700 K were attained, at which time the autocatalytic oxidation reaction resulted in a temperature excursion (at a rate of 10 to 50 K/s) and hydrogen generation. Peak local cladding temperatures are estimated to have exceeded 2600 K, based on information from thermocouples on the outside of the bundle liner. The high-temperature oxidation reaction began at the 2.4- to 3.04-m elevation and formed a localized burn front that moved quickly downward as far as the 1.2-m elevation and then steadily upward. The burn front reached the top end caps (3.80m) and ceased 15 min before the end of the test. The oxidation reaction consumed 75% of the total Zircaloy or almost 100% of the Zircaloy in the path of the burn front. The remaining 25% of the Zircaloy was always below or near the bundle water level. The amount of hydrogen generated was 300±30 g, close to the total conversion of the 1.26-g/s makeup coolant flow within the 45-min high-temperature period. The hydrogen flow fluctuated during the 45-min high-temperature period in response to similar fluctuations (10% to 20% relative) in the bundle coolant flow. The peak hydrogen flow was 190 mg/s, which corresponded to an oxidation power of 28 kW. " FULL-LENGTH HIGH-TEMPERATURE SEVERE FUEL DAMAGE TEST #5 D. D. Lanning N. J. Lombardo W. K. Hensley D. E. Fitzsimmons J. K. Hartwell @ EG&G-Idaho F. E. Panisko April 1988 – Completion Date September 1993 – Publication Date Prepared for U. S. Nuclear Regulatory Commission Under U.S. Department of Energy Contract DE-ACO6-76RLO1830 PNL—6540‖ cited at http://www.osti.gov/energycitations/product.biblio.jsp?query_id=2&page=0&osti_id=10188341 With description: „Post-test visual examination of one side of the fuel bundle revealed no massive relocation and flow blockage; however, rundown of molten cladding was evident." – contradicts to the above description of authors that the cladding burned off above the water level with a rate allowed by water flow. A very typical misrepresentation regarding Zircaloy fires by NRC...

comment #59235 posted on 2013-02-22 16:53:46 by Joel Riddle

As a mechanical engineer who performs nuclear power plant design work, I am curious whether TRACE is publicly available? If so, is it free and could a link be provided here in the comments? Thanks, comment #59216 posted on 2013-02-22 15:44:17 by Zack Hayman aka W Z Hayman III

After watching the fiasco at SONGS, I've been asking "why do main coolant pumps draw suction from the steam generators?" This virtually assures that dry steam pockets will occur. If suction were from the free-flowing RV and discharge to the generators, resistance in the 3/4" tubes would suppress dry steam pockets.

comment #59205 posted on 2013-02-22 15:11:18 by

Scott Krepel, I would like to know if this safety program TRACE is above and beyond or part of the planned VC Sumner's Westinghouse AP1000 unit I have heard and read about ? Regarding the use of the water source for the new VC Sumner reactors, I personally am concerned about the potability of the water used to cool the reactors. From what I have read, the concern for the potability of the drinking water supply appears to be secondary to the safety of the reactors in the case of an accident such as a bursting pipe. James C. Matthews Columbia, SC

comment #65923 posted on 2013-03-05 07:16:11 by The Firestorm Fighter in response to comment #64818

So let us get this straight - you say that NRC does not consider the effects of the auto-catalytic zirconium oxidation during severe accidents, and when asked for specific evidence of why you believe this, you cite for us a technical report about an experiment - sponsored by NRC - designed specifically to help further the NRC's understanding of what happens to fuel rods during severe accidents and gather data for the purpose of improving the models in 2 of the NRC's severe accident codes? Really? I will admit that trying to decipher your logic is quite challenging., but I think I am beginning to see what you are really trying to convey. It sounds like what you want is to have the NRC change the assumptions of the design basis accident so that utilities have to design their plants to prevent or withstand severe accident events. That is quite a different thing than continuing to state that NRC needs to "allow consideration of the effects of the zirconium firestorm". All this time, it sounded like you were simply claiming that the NRC somehow was not, but should be considering the presence or importance of clad oxidation as a heatup mechanism and source of hydrogen during severe accidents - as if to imply their engineers don't somehow know what happens during a severe accident. That seemed ludicrous to me. Oh, one last bit of advice, and I will make this my last words for this thread. You really should tone down on the "firestorm" rhetoric. When you use words that are intentionally chosen so as to inflame, it makes it really hard for reasonable people to take you seriously.

comment #65879 posted on 2013-03-05 01:03:27 by Aladar Stolmar in response to comment #64818

And how I would change the regulations: 1. Perform experiments to verify that a well timed rapid depressurization in the light water (BWR, PWR) reactors can avoid the ignition (ballooning and burst) of cladding of nuclear fuel in the reactor 2. Design means of well timed rapid depressurization and subsequent passive prolonged flooding of the nuclear reactor’s core and deployment of these means 3. Organization of international rapid response team and their national civil defense counterparts for nuclear emergency response coordination 4. Review of channel type reactors to verify the impossibility of development of zirconium fire in the steam (in CANDU), immediate shut down of RBMKs still operating on three sites in Russia 5. Development of evacuation procedures for the time of completion of no2 and perform mock evacuations and public trainings 6. New nuclear power plants should be developed with underground sealed containment systems as per Edward Teller’s suggestion 7. Mandatory release of data collected about the state of reactor during the accident to the public

comment #59263 posted on 2013-02-22 20:36:35 by CaptDCaptD

To: Scott Krepel I urge you to describe how NRC modeling has been able to do anymore that provide educated guesses especially since many other "Experts" have told the NRC that current computer modeling technology cannot provide anything other than information on laboratory "tests" that do not begin to simulate all the variables of the real world! I believe it was Professor Michel Pettigrew that told Commissioner Magwood that same thing at the last NRC San Onofre special meeting held earlier this month in Rockville, MD. Case in point, MHI has built over a hundred SG's and until San Onofre they have all preformed as designed for the most part and they used their own modeling system that has proven successful at least until San Onofre. Others, including AREVA and Westinghouse/Hitachi said that they had no problems with their own modeling, so what we are left with is that the NRC is approving both designs and operational parameters that now leave too much up to chance because they are not making their specific parameters and/or data available and without those, all calculations are suspect. In short, we have been very lucky to date, San Onofre should serve as a wake up call to the NRC that it needs to tighten up its regulations and their enforcement before the USA suffers a nuclear incident or worse a nuclear accident like Fukushima! I would be interested to know if TRACE has been used to model multiple (aka cascade, not just a single tube) of SG tube failures along with a MSLB at San Onofre and if not, why not, since it almost happened for real? Remember Unit 3 had 8 RSG tubes fail in-situ pressure testing along with 1 RSG tube in Unit 2 which was in service (until it was shutdown for refueling) at the same time that had 90% wear, well above the 35% safety limit. If these tube all failed due to a large EQ and or MSLB, it would posed a real threat to San Onofre and all those that live nearby!

comment #63547 posted on 2013-02-27 10:06:50 by richard123456columbia

So they will keep building plants not knowing the risks till they gather data on failures to be plugged into TRACE. This to me is scary that they do not know the potential outcome of these plant designs but go ahead with the risk. They seem to ignore what they do not understand and hope for the best results. Is this proper engineering and design practices? Not in the Engineering school I went to.

comment #64264 posted on 2013-02-28 17:21:14 by CaptD in response to comment #63119

Thank You.

comment #62596 posted on 2013-02-25 12:40:33 by richard123456columbia

Statement: 'Over time, we've put much effort into incorporating existing knowledge into the NRC's thermal hydraulics computer simulation program, TRACE. This program allows NRC staff to construct computer models of the cooling systems of a nuclear power plant and then simulate accidents such as pipe breaks' Until TRACE is perfected, why did they build a plant that is not 100% proven safe, surely they did a mock up of the cooling system to test its use, so what went wrong, is it that it costs too much to do it right or is there no other way of cooling.

comment #59287 posted on 2013-02-22 22:25:13 by richard123456columbia

The nuclear industry had 70+ years and over 400 plants built to make them safe, I see it taken another 100+ or more years to make them safe only if the cost does not prevent it, this industry seems to reduce costs over safety, no one can trust the owners, designers or contractors.

comment #59313 posted on 2013-02-23 00:34:21 by HelpAllHurtNeverBaba

Can MHI ATHOS predict the effects of Fluid Elastic Instability (FEI) and Flashing Feedwater HELB Jet Impingement on the steam generators tubes in SONGS Unit 2, when the steam generator is depressurized due to a main steam line break outside containment along with the failure of MSIV to close. If it can, what is the accuracy of prediction plus minus…..Dr. Pettigrew says it is 20 to 50 percent. Also a 2011 research paper shows that cross-flow fluid velocities at the 90-degree portion on both sides of the u-bend in the hot/cold leg compared with the straight portion of the horizontal leg is double during FEI in large U-bends. This velocity overcomes any resistance provided by the straight flat, curved or honey-comb anti-vibration bars MHI is currently testing in the out-of-plane and/or in-plane direction. SONGS 3 had 8 failed tubes at MSLB test pressures with a vapor fraction of 99.6% and the flat bars failed to prevent the tube-to-tube/AVB wear. Low Contact Forces And Manufacturing Dispersion SCE Theory for SONGS Unit 3 FEI is Bogus and is contradicted/disputed by Westinghouse, AREVA, Dr. Pettigrew, John Large and Bill Hawkins. SCE convinced and sweet-talked MHI in building SONGS large CE Replacement Steam Generators with high heat flows, (MHI was forced to accept numerous untested and unanalyzed design changes to accommodate these high steam flows) in a rush and cheap cost. Now MHI is getting beat by NRC and SCE to cover their mistakes. I am sorry to say, MHI accepted this contract and it is time for MHI to get wise and be out of it rather than try to replace the SONGS steam generators free of cost to please SCE/NRC. MHI in order to maintain their business reputation needs to slow their marketing/manufacturing efforts for US APWR and focus on research/testing/validation in 100% mock-up units before going any further.

comment #59297 posted on 2013-02-22 23:59:44 by HelpAllHurtNeverBaba

Dr. Pettigrew, World's foremost Expert on fluid Elastic Instability stated in 2006, "In nuclear power plant steam generators, U-tubes are very susceptible to undergo fluid elastic instability because of the high velocity of the two-phase mixture flow in the U-tube region and also because of their low natural frequencies in their out of plane modes. In nuclear power plant steam generator design, flat bar supports have been introduced in order to restrain vibrations of the U-tubes in the out of plane direction. Since those supports are not as effective in restraining the in-plane vibrations of the tubes, there is a clear need to verify if fluid elastic instability can occur for a cluster of cylinders preferentially flexible in the flow direction. Almost all the available data about fluid elastic instability of heat exchanger tube bundles concerns tubes that are axisymmetrically flexible. In those cases, the instability is found to be mostly in the direction transverse to the flow. Thus, the direction parallel to the flow has raised less concern in terms of bundle stability." Based on this paper, any knowledgeable Steam Generator (SG) Expert in SG with high heat steam flows such as SONGS, should have designed the steam generators to prevent the adverse effects of fluid elastic instability and flow-induced random vibrations by excluding the following components : 1. Low frequency retainer bars, and 2. Use of flat AVB bars supports to restrain vibrations of the U-tubes in the out-of-plane and in-plane direction. SCE Certified Design Specifications signed by a California Licensed Professional engineer did not specify fluid elastic instability and Mitsubishi did not manufacture steam generators to prevent the adverse effects of fluid elastic instability and flow-induced random vibrations. This could be very well the items, Barbara Boxer is talking about MHI Root cause and MHI/SCE/NRC are covering up under the false pretense of proprietary information. Changing these items or admitting deficiency in these items would have delayed the SGRP Project, was a very expensive change and triggered a NRC Licensing Amendment Process.

comment #59296 posted on 2013-02-22 23:57:03 by HelpAllHurtNeverBaba

Hello Mr. Krepel, Can Trace predict the effects of Fluid Elastic Instability (FEI) and Flashing Feedwater HELB Jet Impingement on the steam generators tubes in SONGS Unit 2 , when the steam generator is depressurized due to a main steam line break outside containment along with the failure of MSIV to close. If it can, what is the accuracy of prediction plus minus.....Dr. Pettigrew says it is 20 to 50 percent. Also a 2011 research paper shows that cross-flow fluid velocities at the 90 degree portion on both sides of the u-bend in the hot/cold leg compared with the straight portion of the horizontal leg is double during FEI in large U-bends. This velocity overcomes any resistance provided by the straight flat or curved bars in the out-of-plane and/or in-plane direction. SONGS 3 had 8 failed tubes at MSLB test pressures with a vapor fraction of 99.6% and the MHI famous flat bars failed to prevent the tube-to-tube/AVB wear. Thanks

comment #59567 posted on 2013-02-23 09:20:45 by Rod Adams in response to comment #59131

@Aladar Stolmar Where do you get the notion that there will be a "zirconium firestorm"? It sounds to me like you have been watching too many of the "typical superhero action movies" that the blog post author mentioned. Science and technology are amazing servants of mankind; they help us to understand the real world, not the imaginary one created by people who do not understand how to perform math or how materials function in action conditions.

comment #64532 posted on 2013-03-01 10:58:10 by CaptD in response to comment #63547

Not in mine or in the one I taught in... I'm looking forward to additional posts from you - Salute! Getting far more qualified people involved and especially professionals from outside of the NRC and most importantly from outside of Region IV, is the first step toward answering basic reactor fatigue safety questions that we now know, affect the entire US Nuclear Fleet. If we learned nothing else from the Fukushima tragedy, we now know that when it come to reactor safety, the widest possible public review can only help insure against future nuclear accidents. Since you are an engineering professional I urge you to read : "Press Release 13-01-22 ATHOS Validity Questioned, Qualifying Investigation Required" Validity of ATHOS computer model requires NRR Qualifying Investigation. (3 Pages) https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=1ltCb57ciXRaOkhK1rhc2BaB0ACXf7MwcSDZZyEAkFDI or this one for much more in-depth technical information: "Response to NRR RAI #32 - Technical" The SCE cannot provide an ACCEPTABLE operational assessment to the NRR, therefore NO RESTART IS POSSIBLE and here ARE THE TECHNICAL REASONS WHY (50 Pages) https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=0BweZ3c0aFXcFX05DMWxKNmZXUTA and/or the even the longer paper: "SCE NRC Presentation analysis + 14 Questions 12-12-17" Technical document includes 14 questions affecting US Reactor SAFETY, that the NRC, NRR and RES Regulators need to ask SCE at their 12/18/12 NRR/RES Meeting. (78 Pages) https://docs.google.com/folder/d/0BweZ3c0aFXcFZGpvRlo4aXJCT2s/edit?docId=0BweZ3c0aFXcFRzBqZUJROWRYNlE

comment #65793 posted on 2013-03-04 16:48:44 by nuclear guy in response to comment #64818

The temperatures you are referring to, 1700K, is roughly 2600 degrees F. This is GREATER than the maximum allowable temperature per 10CFR50.46 (2200F) In other words, the NRC already REQUIRES reactor cores to NEVER EXCEED 2200F during worst case accident conditions. If they never exceed 2200F, they can never exceed 2600F, which is where you consider the start of your "zirconium firestorm". 10CFR50.46 also REQUIRES that no more than 1% of maximum theoritical hydrogen products occur, and that cladding oxidation NOT exceed 17% of maximum. Or in other words, your cited evidence does not call into question any deficiencies which can be readily observed in quantified ECCS performance data in US regulations. Also, the US NRC had nothing to do with the Fukushima disaster, despite your somewhat unusual comment. Fukushima was in Japan, the US NRC does not govern Japan. Fukushima also was not caused by any "zirconium firestorm". It was caused by a loss of decay heat removal. I think you should just reword your entire argument to "I don't like solid fuel nuclear power", rather than try to construe some technical non-sense to make it sound like there is some big unknown issue that nuclear power experts have been "ignoring" for over 70 years.

comment #63120 posted on 2013-02-26 17:17:19 by Moderator in response to comment #59296

TRACE is a systems analysis code, so it is not used to predict FEI or structural failure. So the issues you discuss are not addressed by TRACE. Scott Krepel

comment #63119 posted on 2013-02-26 17:14:53 by Moderator in response to comment #59263

The assessment database used to determine TRACE's validity includes "separate effects" tests and "integral effects" tests. The former are experiments that are performed to isolate specific thermal hydraulic phenomena and develop correlations for a variety of different conditions. The latter are run at scaled test facilities that represent specific nuclear plant designs, including all of the relevant systems. These integral effects tests are simulations of accident scenarios, which can be used to develop confidence that TRACE is adequately predicting relevant events during different plant scenarios. The design and operational parameters for specific plants can then be used to generate plant-specific models and investigate accident scenarios. The intent of TRACE is to provide NRC staff with an independent tool to assess applicants' analysis results. The manuals that describe the modeling and primary uses of TRACE can be found in NRC's public ADAMS system under ML120060403, including the assessment manuals that describe various assessments performed on TRACE. I am not currently doing work associated with San Onofre, so I can't answer these questions. This type of concern would probably be better addressed by the San Onofre review board. As such, they have been passed along, but all members of the public are encouraged to raise any concerns during public hearings, public comment periods, and other opportunities that the NRC provides to solicit feedback. Scott Krepel

comment #63115 posted on 2013-02-26 17:11:01 by Moderator in response to comment #59235

Information on how to obtain TRACE can be found here: http://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html . Scott Krepel

comment #63114 posted on 2013-02-26 17:10:22 by Moderator in response to comment #59205

TRACE is a systems analysis code, and is not used to determine the potability of drinking water near nuclear plants. Scott Krepel

comment #63113 posted on 2013-02-26 17:06:41 by Moderator in response to comment #59313

TRACE is a systems analysis code, so it is not used to predict FEI or structural failure. So the issues you discuss are not addressed by TRACE. Scott Krepel comment #62953 posted on 2013-02-26 11:42:54 by Aladar Stolmar in response to comment #59567

The key process in 1979 TMI-2 accident, the 1986 Chernobyl-4 accident, the 2003 Paks 2 refueling pond washing vessel incident and the Fukushima Daiichi 1, 2 and 3 reactors March 2011 accidents. A few of us, nuclear engineers were, are fighting for lifetime for the consideration of real processes in the reactor severe accidents. As I formulated in a comment to US NRC: Consideration of the zirconium-steam reaction and the ignition and intense firestorm in nuclear reactor fuel rods is well overdue. Reevaluating the evidence provided by the TMI-2 reactor accident, Chernobyl-4 reactor accident, and Paks Unit 2 fuel washing incident, (Fukushima Daiichi units 1-4 fuel damage) with consideration of this intense fiery process, will bring us closer to an ultimately safe nuclear power plant design. http://pbadupws.nrc.gov/docs/ML1033/ML103340250.pdf Also, I called two years ago for a review: If the hydrogen which is generated in the reactor core from the reaction of the steam (coolant) with the zirconium alloy (or other low neutron absorbing metal cladding and other fuel bundle elements) explodes inside the building surrounding the reactor, this detonation still will not cause a break of the pressure boundary of the containment. Thirty years after the TMI-2 accident and 23 years after the Chernobyl disaster, I feel obligated to formulate this guideline in order to protect the public from further irradiation from the use of nuclear power. The Chernobyl type reactors (RBMK), which are still operating, have to be shut down immediately because they do not satisfy this guideline. Other nuclear reactors operating and future designs shall be reviewed for compliance to this key requirement and the result of such review shall be defining for their future. http://aladar-mychernobyl.blogspot.com/ Returning to the comment to US NRC http://pbadupws.nrc.gov/docs/ML1033/ML103340250.pdf : „It is a much overdue duty of NRC and IAEA to evaluate the evidence provided by the TMI-2 accident, Chernobyl-4 accident, Paks-2 incident, and related experiments. Evaluating this evidence, one can see that the ignition of the zirconium fire in the steam occurs at a local temperature of the fuel cladding of around 1000-1200'C, [[and that a self-feeding with steam due to the precipitation of eroded fuel pellets and zirconia reaction product from the hydrogen stream into the water pool, causes intense evaporation.]] There are insignificant differences in the progression of the firestorms that occurred in the TMI-2 reactor severe accident, Paks washing vessel incident, and Chernobyl-4 reactor accident; the later defined only by the amount of zirconium available for the reaction. At the mean time, there are significant similarities in the processes leading to the ignition of the firestorm. In all three of the compared cases, it took several hours of ill-fated actions or in-actions of the operators to cause the ignition condition. Also, there are similarities in the end result of the firestorm; namely, that the extent of the fuel damage is much less than it was predicted from any other severe fuel damage causing scenarios, introduced for explanations. Therefore the fraction of released fission products is significantly less than was anticipated from the fuel melting or a so called "steam explosion" scenario. Also, the fiery steam-zirconium reaction results in a much higher than anticipated (from any other scenarios) rate of Hydrogen production, which in turn requires a review of containment designs."

Celebrating African American History Month: NRC Applauds Achievement of Dr. Haile K. Lindsay

posted on Tue, 26 Feb 2013 15:37:42 +0000

Note: Dr. Lindsay was honored this month with a Special Recognition Award at the 2013 Black Engineer of the Year Science, Technology, Engineering, and Mathematics (STEM) Conference. Lindsay holds a B.S., M.S., and Ph.D. in mechanical engineering from North Carolina A&T State University.
Haile K. Lindsay
Thermal Engineer

After receiving my award, I was asked to write a bit about what I do at the NRC and how I contribute to the African American community –

thus this post. I will have been with the agency for five years. I came to the NRC right after getting my PhD. My job is to review the thermal and containment sections of the license applications we receive for packages to either store or transport spent fuel or radioactive materials. My job allows me to apply the knowledge I acquired in school about heat transfer, thermodynamics, and other critical subjects. I review the package designs to see if they meet NRC regulations for safety and security of people and the environment. If a design does not meet our requirements, we will not issue a license. The mantra I live by is: "Hard Work Pays Off." If you work hard, you can be successful at anything you do. I saw that come true as a student, and now in my career at the NRC. My dissertation focused on treating liposarcoma (a rare tumor that develops in fat cells, typically in extremities) using hyperthermia – that is, heating cancer cells enough to destroy them. I am proud that my research contributed to the body of knowledge on this relatively new mode of cancer treatment. At work, I was honored to learn that my branch chief at the time, Victor Cusumano, had nominated me for this prestigious award. I credit my hard work toward becoming a qualified thermal reviewer, ensuring that corporate knowledge is transferred to the newer NRC employees, as well as papers I have presented at nuclear and government conferences and the work I have done to organize NRC conferences. I also work hard to give back to my community. In my role as NRC Chapter President for Blacks in Government, last year I organized a clothing and toiletries drive entitled Winter H.O.P.E. (Helping Others by Providing Them Essentials). The clothing and toiletries, donated by the NRC staff, were given to The Dwelling Place – an organization that provides housing opportunities and support services in Montgomery County, Md., for families experiencing homelessness. I also organized a luncheon last summer for 17 D.C. Summer Youth Employment Program interns who worked at the NRC. We provided pizza and organized a panel to talk with these young people about our respective career paths and provide some helpful tips for success. I feel truly honored and blessed to have been recognized with such a prestigious award. This award will fuel me to continuously do even greater work for this agency and my community.

Comments

comment #63520 posted on 2013-02-27 09:16:13 by Beth Deahl

Congratulations, Dr. Lindsay.

comment #63045 posted on 2013-02-26 14:52:37 by LaTosha

This is great and I am so proud of your wonderful achievement.

Deconstructing the Decommissioning Process

posted on Thu, 28 Feb 2013 16:09:09 +0000

Dave McIntyre
Public Affairs Officer

Duke Energy's decision to shut down the Crystal River 3 reactor in Florida rather than pay for expensive repairs to its containment dome has focused attention once more on the lengthy process for decommissioning nuclear power plants. Since Dominion Nuclear's announcement last year that it will shutter its Kewaunee plant in Wisconsin, the country now has two reactors entering this process. Duke took the first step on

Feb. 20, when it gave the NRC its official certification that it had permanently ceased operations at Crystal River 3 and permanently removed the fuel from the reactor. Those certifications effectively changed the plant's operating license to a "possession only" license – in other words, the company is no longer permitted to load fuel into the reactor vessel and operate the plant. After these initial certifications, the process can be quite slow. Duke will have up to two years to develop and submit its decommissioning plan – officially called the post-shutdown decommissioning activities report, or PSDAR in NRC-speak. The report will include a description and schedule for decommissioning activities, their estimated cost, and a discussion of why any anticipated environmental impacts have already been reviewed in previous environmental reports on the plant. Once the NRC receives the PSDAR, we will publish it for public comment and conduct a public meeting near Crystal River to explain the decommissioning process. Duke will not be able to conduct any major decommissioning activities until 90 days after NRC receives the PSDAR. Under NRC regulations, Duke can take up to 60 years to complete the process, from cessation of operations to final decommissioning and termination of license. Why so long? There are actually two advantages: Radioactivity decays over time, making the final cleanup easier; and the company's decommissioning trust fund continues to grow. This stage of decommissioning is called SAFSTOR, as the company maintains the shuttered plant in safe storage until final cleanup begins. Throughout this process, Duke will be able to use some of its decommissioning funds. It can spend up to 3 percent of the fund on decommissioning planning as it develops the PSDAR, and up to 20 percent to maintain and monitor plant safety during the SAFSTOR period. NRC limits use of the funds to ensure that enough money remains to complete cleanup and follow the process through to license termination. The NRC requires Duke to clean up the site so that residual radiation is quite low – specifically, that no person on the site would receive a dose above 25 millirem per year. (In comparison, the average American receives 310 millirem per year from natural radiation, and the dose from a single chest X-ray is about 10 millirem.) At least two years before Duke reaches that point, it must submit a license termination plan, detailing the final steps. NRC inspectors will verify that the site has been decontaminated to the NRC's requirements. Duke will then ask the NRC to terminate the license, or modify it to apply only to a spent fuel storage facility, if needed. One popular question is the cost of decommissioning a nuclear power plant. Estimates vary, and of course it has been several years since a plant has been decommissioned, but the NRC estimates that the costs generally range from $300-400 million. This estimate applies only to NRC-mandated activities – in other words, reaching the radiological criteria. Dismantling other parts of the plant (such as support buildings) would cost extra, so the company's estimate might be higher. Throughout the entire decommissioning process, the NRC's objective is to protect public health and safety while ensuring that the site is cleaned up to our requirements. For more information, NRC regulations on decommissioning are 10 CFR 50.82 and 10 CFR 20, Subpart E. Additional Note: There are two other possible methods for decommissioning. DECON involves active decontamination of the site, either immediately after operations cease or after a period of SAFSTOR. The third, ENTOMB, is just what it sounds like - radioactive contaminants are permanently encased on site in structurally sound material such as concrete and appropriately maintained and monitored until the radioactivity decays to a level permitting restricted release of the property. To date, no NRC-licensed facilities have requested the ENTOMB option.

Comments

comment #64585 posted on 2013-03-01 13:05:36 by Moderator in response to comment #64294

The owners of these two sites made this business decision to decommission these sites. Any decision to pursue a restart in the future or use the site as a museum, once it is safe to do so, would be a decision by the respective owners. Dave McIntyre

comment #64294 posted on 2013-02-28 19:55:10 by BobinPgh

I just think it is a terrible waste that kewauness has nothing wrong with it and it has to be decommissioned. It is possible to have something between full operation and decomissioning that would make it possible to shut down a reactor temporarily until is may be needed again? That is, would it be possible to preserve the parts inside without having to take it down? Or would some taking apart be done during safestor? I always wondered if it might be possible to make a no longer used power station a national nuclear power museum. Would an idea like this be possible?

comment #64194 posted on 2013-02-28 11:18:35 by CaptD

Where are the highly radioactive materials going to be stored once they are removed from the site? IMO This is why we are now seeing the time to decommission stated as 60+ years... All these old reactors sites will simply become nuclear material storage sites.

comment #64231 posted on 2013-02-28 14:36:09 by Moderator in response to comment #64194

There are several former reactor sites where decommissioning has been completed but spent fuel remains onsite. In these cases the operating license is modified to cover the fuel storage facility, while the rest of the site can be released for public use. The fuel is still there because the federal government has not yet developed a permanent disposal site like the one that was envisioned for Yucca Mountain. Last year, the Blue Ribbon Commission on America's Nuclear Future recommended establishing some centralized, regional storage sites for spent fuel, with a priority on removing the stored fuel from the decommissioned reactor sites. Last month, the Energy Department endorsed that strategy, but of course licensing and establishing such a site will take time. It is impossible to predict right now whether such a site or even a permanent disposal option might be available by the time Crystal River 3 and Kewaunee complete their decommissioning. The 60-year period for decommissioning really is unrelated to this issue. The time limit was included in NRC regulations for decommissioning long before the lack of disposal capacity for spent fuel became a problem.
Dave McIntyre