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April 16, 2018

Hubert Bell, Inspector General
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2728

SUBJECT: Inexplicable NRC Handling of BWR Safety Limit Problem

Dear Mr. Bell:

On behalf of the Alliance for a Green Economy, Beyond Nuclear, the Citizens' Environmental Coalition, Don't Waste Michigan, the Nuclear Information and Resource Service, and the Union of Concerned Scientists, I respectfully ask the Office of the Inspector General to look into how the Nuclear Regulatory Commission staff handled the boiling water reactor (BWR) safety limit problem that was reported to the NRC by General Electric (GE) in March 2005. All the boiling water reactors operating in the United States were potentially affected by this problem according to GE. Basically, GE informed the NRC that the computer analyses it performed of design bases transients had assumed an automatic reactor scram would terminate the transient before one of four BWR safety limits was violated, but more recent analysis showed that the reactor scram would not happen during one depressurization transient in time to prevent violating the safety limit.

As detailed in the attached report, the owners of these reactors initially attempted to resolve the problem by revising the technical specification bases for the safety limit to indicate that it was permissible to momentarily violate the safety limit. The NRC considered this request and denied it in writing. The NRC staff wrote in its denial that it was "*concerned that in some depressurization events which occur at or near full power, there may be enough bundle stored energy to cause some fuel damage*."

Yet despite knowing about a problem that could result in fuel damage, the NRC has taken over a decade to resolve this problem. And the solution has only been implemented at 28 of the nation's 34 operating boiling water reactors. While two of those reactors use a different fuel type (i.e., not GE fuel) and may not be susceptible to the problem, it's not clear why the other four reactors would not also require the fix to this problem. The most recent BWR to take the cure—Nine Mile Point Unit 2—required a significant change to a setpoint that triggered automatic actions sooner to protect against fuel damage, so the fix is more than mere computer re-analysis to straighten out a paperwork glitch.

We feel it is imperative that OIG expeditiously get the NRC to answer, on the record, the seven questions posed on pages 9 and 10 of the attached report for two reasons. First and foremost, there may be reactors operating today with a problem that could result in fuel damage should a depressurization event occur. Since the solution has been established for nearly a decade, there's no justification for any reactor to operate at elevated and undue risk to the people living nearby. Second, the NRC is currently looking to develop transformational methods that accelerate deployment of industry's business innovations like digital instrumentation and control systems. The NRC should not build an express lane for nuclear business desires when it permits known safety problems like this one to languish uncorrected for over a decade in parking lot. Something seems to have gone dreadfully wrong here, and the answers to the seven questions raised in the report could help the NRC avoid wrongs in the future, or at least lessen their dreadfulness.

Sincerely,

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The Boiling Water Reactor Safety Limit Problem

The Safety Limit Problem Reported

On March 29, 2005, General Electric (GE) reported a problem to the Nuclear Regulatory Commission (NRC):

> This letter provides information concerning a condition that GE has determined to be reportable under 10CFR21, even though it does not produce a substantial safety hazard. As a result of improvements in calculation methods, GE has identified an anticipated operational occurrence (AOO), the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient, that could result in a condition in which Safety Limit (SL) 2.1.1.1 may be exceeded. Depending upon the plant-specific response to a PRFO, reactor steam dome pressure could decrease to below 785 psig while thermal power exceeds 25% of rated, which would be a violation of SL 2.1.1.1.

The Safety Limit Problem Translated

10CFR21 (Title 10 of the Code of Federal Regulations, Part 21) is a federal regulation that requires companies providing products or services to nuclear power plants to notify the NRC of defects that could create a substantial safety hazard. GE designed all the boiling water reactors (BWRs) now operating in the United States and provides many BWRs with fuel.

Other federal regulations require that nuclear plants be designed to minimize the chances that nuclear fuel becomes damaged during anticipated operational occurrences (AOOs). The regulations define an anticipated operational occurrence as being "*conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.*" The AOOs vary depending on the type of reactor design. For BWRs, the AOOs include the pressure regulator failure—maximum demand (open) (PRFO).

In BWRs, steam is produced within the reactor vessel as the energy produced when atoms split heats water flowing through the reactor core to the boiling point. Steam leaves the reactor vessel and flows through pipes that pass through the primary containment wall into the turbine building. The pressure regulator determines where the steam goes. When the turbine/generator is offline (i.e., when the reactor is shut down or at power levels less than about 15 percent, the turbine's stop valves (labeled SV in Figure 1) are closed. When the steam pressure rises above the setpoint specified by the operators (typically around 950 pounds per square inch), the pressure regulator opens the bypass valves (labeled BPV) to pass the steam directly into the main condenser. When the turbine/generator is online, the pressure regulator closes the bypass valves[1] and adjusts the position of the turbine's control valves (labeled CV). The pressure regulator responds to reactor power level changes by opening or closing the control valves as necessary to maintain the pressure at the inlet to the turbine at the setpoint value.

The PRFO event postulates that an input to the control system or a circuit card within the control system fails such that the system sends out signals causing all the control valves and bypass valves to open fully. The rapid opening of these valves allows steam to flow more readily through the turbine as well as directly into the main condenser. The higher steam flow reduces the pressure inside the reactor vessel. The decreasing reactor vessel pressure causes its water level to rise, similar to how the fluid level inside a soda bottle that has been shaken rises when its cap is loosened (but for different physical reasons).

The reactor protection system for BWRs monitors key plant parameters and automatically triggers rapid insertion of the control rods into the reactor core to stop the nuclear chain reaction upon apparent signs of

[1] The bypass valves can handle about 30 percent of the rated steam flow. Thus, the turbine/generator needs to be placed online before the reactor power level exceeds 30 percent; otherwise, the bypass valves will be fully open and be unable to handle the "extra" steam flow.

trouble. High water level in the reactor vessel and closure of the main steam isolation valves (labeled MSIV in Figure 1) are two of the conditions that trigger automatic shutdown.[2] The PRFO event starts a race between rising water level and decreasing reactor vessel pressure as to which condition will automatically shut down the reactor first.

The winner of this race for postulated PRFO events is determined by computer analyses. Computer analyses are performed of design bases transients (i.e., the AOOs) and accidents (e.g, pipe breaks) to demonstrate that the reactor's response will be within safety boundaries approved by the NRC when they licensed the reactor to operate. GE's computer models for AOOs evolved over time, from the REDY code through the ODYN code to the TRACG code. GE's early analyses assumed that the reactor vessel level rise caused by depressurization during a PRFO event would win the race and trigger an automatic reactor shutdown. The latest GE analyses suggested that the reactor vessel pressure could drop below the setpoint that automatically closes the MSIVs when the reactor is operating at high power levels before the water level in the reactor vessel rises high enough to trigger automatic reactor shutdown.

The prediction of a different winner for the race to shut down the reactor during a PRFO event is important because if high vessel water level does not win the race, a BWR Safety Limit could be violated. There are only four BWR Safety Limits as shown below: three reactor core safety limits and one reactor coolant system pressure safety limit. GE notified the NRC that its latest analytical methods revealed that Safety Limit 2.1.1.1 could be violated during a PRFO event because reactor vessel pressure could drop below 785 pounds per square inch (psig) before the reactor power level dropped below 25 percent. When high vessel water level wins the race, the rapid insertion of the control rods reduces the reactor power level below 25 percent before reactor vessel pressure drops below 785 psig.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ [1.07] for two recirculation loop operation or ≥ [1.08] for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

Safety Limit 2.1.1.2 requires the Minimum Critical Power Ratio or MCPR, to be satisifed when the reactor vessel pressure is above 785 psig. MCPR is one of three thermal limits guarding against fuel damage. It protects against fuel damage during AOOs. The other two thermal limits protect against fuel damage during steady state operation and during postulated accidents. The MCPR limit prevents the

[2] High water level in the reactor vessel triggers automatic closure of the turbine stop and control valves, which in turn triggers the automatic shutdown of the reactor when the power level is above about 30 percent (i.e., the bypass valve flow capacity.)

power produced by individual fuel bundles from exceeding the critical power above which the impaired ability to remove the heat could cause overheating damage during an AOO. See Appendix 1 for a fuller explanation of this fuel thermal limit and the damage that can result when it is violated.

GE conducted experiments in laboratories that simulated a fuel bundle and its cooling to collect data on heat transfer properties. Computer codes used this data to model actual laboratory test conditions and then postulated reactor operating conditions. The experiments upon which the computer codes were developed were not conducted at high power/low pressure conditions. Therefore, Safety Limit 2.1.1.1 supports Safety Limit 2.1.1.2 by preventing the reactor from operating above 25 percent power when the reactor vessel pressure is below 785 psig, or in an operating regime not studied in experiments and not modeled by the computer codes.

The NRC explained the role played by these two Safety Limits:

> GDC 10 [General Design Criterion 10, a federal regulation] requires, and SLs [safety limits] ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2
>
> The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling. Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

One BWR owner stated that the intent of Safety Limit 2.1.1.1 is to prevent the reactor from operating in a region where the MCPR analysis is invalid. In other words, the safety analyses required by the regulations provide reasonable assurance that fuel damage will not occur during AOOs initiating from within the analytical boundaries. If an AOO event occurs outside the boundaries, fuel damage may occur.

The potential consequence from violating Safety Limit 2.1.1.1 is explained by the NRC's Enforcement Policy which features two processes. One process classifies violations of regulatory requirements into four color-coded bins (green, white, yellow, and red in order of increasing safety significance.) The other process, termed Traditional Enforcement, classifies violations into four numerical Severity Levels:

- **SL I violations are those that resulted in or could have resulted in serious safety or security consequences (e.g., violations that created the substantial potential for serious safety or security consequences or violations that involved systems failing when actually called on to prevent or mitigate a serious safety or security event).** [emphasis added]

- SL II violations are those that resulted in or could have resulted in significant safety or security consequences (e.g., violations that created the potential for substantial safety or security consequences or violations that involved systems not being capable, for an extended period, of preventing or mitigating a serious safety or security event).

- SL III violations are those that resulted in or could have resulted in moderate safety or security consequences (e.g., violations that created a potential for moderate safety or security consequences or violations that involved systems not being capable, for a relatively short period, of preventing or mitigating a serious safety or security event).
- SL IV violations are those that are less serious, but are of more than minor concern, that resulted in no or relatively inappreciable potential safety or security consequences (e.g., violations that created the potential of more than minor safety or security consequences).

The Enforcement Policy provides three examples of Severity Level I violations:

> 6.1 Reactor Operations
>
> a. *SL I* violations involve, for example:
>
> 1. A system[10] that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident (DBA) or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier is unable to perform its licensing basis safety function[11] when actually called on to function;
> 2. An inadvertent or unplanned criticality; or
> 3. A technical specification safety limit is exceeded.

Violating a technical specification safety limit, such as Safety Limit 2.1.1.1, is explicitly identified among the trio of transgressions warranting a Severity Level I sanction. In other words, violating a Safety Limit is more like a felony than a misdemeanor. This would imply that reactors known to be operating with the potential for violating a safety limit would have that condition resolved expeditously. But that simply did not happen.

The Safety Limits Problem's Solution – First Attempt

The Safety Limits are part of the technical specifications issued as an appendix by the NRC to each reactor operating license. Plant owners can submit license amendment requests to the NRC seeking changes to the technical specifications. But only the NRC can approve changes.

The Technical Specifications Task Force (TSFT) is an industry group consisting of representatives from the owners groups for the four reactor types: the Boiling Water Reactor Owners Group (BWROG), the Pressurized Water Reactor Owner Group/Westinghouse (PWROG/W), the Pressurized Water Reactor Owner Group/Combustion Engineering (PWROG/CE), and the Pressurized Water Reactor Owner Group/Babcock & Wilcox (PWROG/B&W). On July 18, 2006, the TSTF submitted a request to the NRC to revise the bases for the Safety Limits in the BWR standardized technical specifications. The bases accompany the technical specifictions and explain the safety role played by the limits and conditions within the technical specifications. The TSTF conceded that Safety Limit (SL) 2.1.1.1 could be violated during a PRFO event, but proposed revising the bases:

> Therefore, the SL 2.1.1.1 Bases are revised to state that the safety limit is not intended to apply to depressurization transients, such as PRFO, that may result in momentarily decreasing below 785 psig with thermal power above 25%, but which do not result in steady state operation under those conditions.

BWR Safety Limit Problem

The NRC reviewed the TSFT request and denied it in a letter adated August 27, 2007. The NRC explained its reasons for denying the request:

> Standard Technical Specifications, Section 5.5.14(b)(1), "Technical Specifications (TS) Bases Control Program," states that licensees may make changes to Bases without prior NRC approval, provided the changes do not involve a change in the TS incorporated in the license. The proposed change to the TS Bases has the effect of relaxing, and hence, changing, the TS Safety Limit. An exception to a stated TS safety limit must be made in the TS and not in the TS Bases. In addition, a potential exists that the requested change in the TS Bases could have an adverse effect on maintaining the reactor core safety limits specified in the Technical Specifications, and thus, may result in violation of the stated requirements. Therefore, from a regulatory standpoint, the proposed change to the TS Bases is not acceptable.
>
> and
>
> … the staff is concerned that in some depressurization events which occur at or near full power, **there may be enough bundle stored energy to cause some fuel damage**. If a reactor scram does not occur automatically, the operator may have insufficient time to recognize the condition and to take the appropriate actions to bring the reactor to a safe configuration. [emphasis added]

So, the NRC denied the request by TSTF because it was legally unacceptable and technically unsound, two pretty good reasons.

The BWROG gave up on a generic resolution of Safety Limit problem in April 2012 and recommended that owners of U.S. boiling water reactors submit license amendment requests to fix the problem. The license amendment requests would revise the reactor vessel pressure value in Safety Limit 2.1.1.1 to bound the pressure range used in the computer studies that established the MCPR limits in Safety Limit 2.1.1.2. In other words, the license amendment requests would restore assurance that BWRs could experience a PRFO event without it causing fuel damage.

BWR Safety Limit Problem

The Safety Limits Problem's Partial Solution

Eigth months after the BWROG recommended that owners submit license amendment requests to the NRC to resolve the safety limit problem, **7.8 years** after GE notified the NRC about the problem, and **5.3 years** after the NRC denied the request the resolve the problem via a revision to the generic technical specification bases, the owner of the two boiling water reactors at the Susquehanna Steam Electric Station in Pennsylvania applied to the NRC for a revision to the operating licenses to correct the problem. Over the following four years, owners of 26 other U.S. BWRs submitted license amendment requests to the NRC to correct the problem.

Table 1: License Amendments to Resolve BWR Safety Limit Problem

Reactor	License Amendment Request	License Amendment	Original Reactor Pressure	Revised Reactor Pressure
Susquehanna Units 1 and 2	12/31/2012	12/08/2014	785 psig	557 psig
Monticello	03/11/2013	11/25/2014	785 psig	686 psig
Pilgrim	04/05/2013	03/12/2015	785 psig	685 psig
River Bend	05/28/2013	12/11/2014	785 psig	685 psig
FitzPatrick	10/08/2013	02/09/2015	785 psig	685 psig
Hatch Units 1 and 2	03/24/2014	10/20/2014	785 psig	685 psig
Browns Ferry Units 1, 2, and 3	12/11/2014	12/16/2015	785 psig	585 psig
Duane Arnold	08/06/2015	08/18/2016	785 psig	686 psig
Clinton	08/18/2015	05/11/2016	785 psig	700 psia
Dresden Units 2 and 3	08/18/2015	05/11/2016	785 psig	685 psig
Quad Cities Units 1 and 2	08/18/2015	05/11/2016	785 psig	685 psig
LaSalle Units 1 and 2	11/19/2015	08/23/2016	785 psig	700 psia
Peach Bottom Units 2 and 3	12/15/2015	04/27/2016	785 psig	700 psia
Limerick Units 1 and 2	01/15/2016	11/21/2016	785 psig	700 psia
Columbia Generating Station	07/12/2016	06/27/2017	785 psig	686 psig
Nine Mile Point Unit 1	08/01/2016	11/29/2016	785 psig	700 psia
Oyster Creek	08/01/2016	11/29/2016	785 psig	700 psia
Perry	11/01/2016	06/19/2017	785 psig	686 psig
Nine Mile Point Unit 2	12/13/2016	10/31/2017	785 psig	700 psia
Brunswick Units 1 and 2	None found	None found	785 psig	Not revised
Cooper	None found	None found	785 psig	Not revised
Fermi Unit 2	None found	None found	785 psig	Not revised
Grand Gulf	None found	None found	785 psig	Not revised
Hope Creek	None found	None found	785 psig	Not revised

But there are 34 BWRs operating in the United States. As shown in Table 1, six BWRs operating in 2018 have not obtained the solution recommended by the BWROG in April 2012 to the Safety Limit problem reported to the NRC by GE in March 2005.

The solution sometimes involved more than merely revising the reactor pressure value in Safety Limit 2.1.1.1 to reconcile it with the pressure range assumed in the computer analyses. For example, the solution for Nine Mile Point Unit 2—the most recent BWR to get fixed—also entailed raising the setpoint at which low pressure triggers automatic closure of the main steam isolation valves which in turn triggers automatic shut down of the reactor to terminate the transient from 746 pounds per square inch to 814 pounds per square inch. This setpoint change means that the reactor will automatically trip at a higher reactor pressure, and therefore sooner, during a PRFO event. Thus, more than mere re-analysis was needed to prevent the safety limit from being violated—a physical change that tripped the reactor more rapidly was also required.

The Safety Limits Problem's Solution Glacial[3] Pace

Table 2 shows how long it took to resolve the safety limit problem at the nation's BWRs. It took an average of 10.9 years from when GE reported the problem to the NRC to the time the NRC issued an amendment to resolve it. It took an average of 8.5 years from the time when NRC denied the request to resolve the problem via a revision to the technical specification bases to the time when the NRC issued an amendment to resolve it. And these averages exclude the ticking clocks for the six BWRs that may still be affected by the problem.

Table 2: Time ~~Needed~~ Taken to Resolve BWR Safety Limit Problem

Reactor	GE Report to Amendment	NRC's Denial to Amendment
Susquehanna Units 1 and 2	9.7 years	7.3 years
Monticello	9.7 years	7.2 years
Pilgrim	10.0 years	7.5 years
River Bend	9.7 years	7.3 years
FitzPatrick	9.9 years	7.5 years
Hatch Units 1 and 2	9.6 years	7.1 years
Browns Ferry Units 1, 2, and 3	10.7 years	8.3 years
Duane Arnold	11.4 years	9.0 years
Clinton	11.1 years	8.7 years
Dresden Units 2 and 3	11.1 years	8.7 years
Quad Cities Units 1 and 2	11.1 years	8.7 years
LaSalle Units 1 and 2	11.4 years	9.0 years
Peach Bottom Units 2 and 3	11.1 years	8.7 years
Limerick Units 1 and 2	11.6 years	9.2 years
Columbia Generating Station	12.2 years	9.8 years
Nine Mile Point Unit 1	11.7 years	9.3 years
Oyster Creek	11.7 years	9.3 years
Perry	12.2 years	9.8 years
Nine Mile Point Unit 2	12.6 years	10.2 years
Average Resolution Time	**10.9 years**	**8.5 years**
Brunswick Units 1 and 2	13.0 years and counting	10.6 years and counting
Cooper	13.0 years and counting	10.6 years and counting
Fermi Unit 2	13.0 years and counting	10.6 years and counting
Grand Gulf	13.0 years and counting	10.6 years and counting
Hope Creek	13.0 years and counting	10.6 years and counting

The Safety Limits Problem's Missing Solution

As shown in Tables 1 and 2, six of the 34 boiling water reactors operating in the U.S. have not pursued the fix recommended by the BWROG in April 2012 for the safety limit problem reported by GE in March 2005.

There are reasons to believe that the GE analytical problem may not affect Brunswick Units 1 and 2 for the simple fact that these reactors use Areva's ATRIUM 10XM and 11 fuel rather than fuel provided by

[3] Apologies to glaciers for calling out their slow pace. Glaciers do not have feet and arms. Lacking means to propel themselves, glaciers must rely on gravity to move them along. The NRC has many feet and arms, but seems unable to recognize the gravity of the unresolved safety limit problem.

GE. And Grand Gulf as Global Nuclear Fuel fuel which may or may not be affected by GE's analytical error.

Even if valid, this reason does not seem to apply to Cooper which had and still has GE fuel in its reactor core. Or to Fermi Unit 2 and Hope Creek which also have GE fuel in its reactor core.

So, at least three and perhaps six BWRs that had the potential for violating a Safety Limit during a PRFO event have not implemented the solution adopted at the other 28 BWRs.

The Safety Limits Problem's Missed Opportunities

It's clear there were many opportunities to have corrected this Safety Limit problem long ago. It's not clear why these many opportunities failed to result in solutions.

For example, Hope Creek is a BWR that has yet to have Safety Limit 2.1.1.1 revised as has happened at 28 other BWRs. But Safety Limit 2.1.1.2 has been revised at Hope Creek since April 2012 when the BWROG recommended license amendment requests to fix Safety Limit 2.1.1.1. On June 8, 2016, its owner submitted a license amendment request to the NRC seeking to revise Safety Limit 2.1.1.2 to reflect the next operating cycle's MCPR limit. The NRC issued the amendment on October 13, 2015 and revised the MCPR limit to 1.08 or greater when both recirculation pumps are running and 1.11 or greater when only one recirculation pump is running:

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 24% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 24% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow:

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be ≥ 1.08 for two recirculation loop operation and shall be ≥ 1.11 for single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow and the MCPR below the values for the fuel stated in LCO 2.1.2, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

But the reactor pressure value in Safety Limit 2.1.1.1 and 2.1.1.2 was left unchanged at 785 psig. As shown in Table 1, 785 psig is the reactor pressure that got reduced at 28 BWRs with GE fuel to resolve the Safety Limit 2.1.1.1 problem. It remains at 785 psig at Hope Creek. So, this amendment incorporated the current MCPR limit. In other words, it specified the corrected MCPR limit over the uncorrected pressure range.

BWR Safety Limit Problem

And Hope Creek is not an isolated case. It happened many times at many BWRs between April 2012 when the BWROG recommended the fix and when the NRC issued the amendment to correct Safety Limit 2.1.1.1 (or today for BWRs that have yet to seek and obtain this solution.)

For example, the owner of the LaSalle Unit 2 reactor submitted a license amendment request to the NRC on October 11, 2012—eight months after the BWROG recommendation—to revise Safety Limit 2.1.1.2 to incorporate the proper MCPR limit. The NRC issued the amendment on February 27, 2013. Like at Hoep Creek, LaSalle then had the proper MCPR limits for the improper reactor pressure ranges. LaSalle's owner waited until November 19, 2015, to submit a license amendment request to the NRC to correct the Safety Limit 2.1.1.1. error. And the NRC did not issue the associated amendment until August 23, 2016.

The Safety Limit problem was identified in March 2005. The NRC denied the industry's paper fix in August 2007. The BWROG recommended a license amendment request solution path in April 2012. Yet despite having both the problem and its solution well established, LaSalle's owner submitted a license amendment request in October 2012 that ignored the matter altogether. And the NRC approved the request to impose the correct MCPR limit associated with the known-to-be-wrong reactor pressure limit in February 2013. More than two years pass before LaSalle's owner submitted the license amendment request to the NRC to resolve a Safety Limit problem identified more than a decade earlier.

The Safety Limits Problem's Unanswered Questions

The Safety Limit problem reported by GE in March 2005 and slowly resolved by the many, but not all, of the potentially affected BWRs raises several questions:

1. In its denial of the Technical Specifications Task Force proposal to resolve the problem via a revision to the bases for the Safety Limit, the NRC staff stated its concern "*that in some depressurization events which occur at or near full power, there may be enough bundle stored energy to cause some fuel damage*." In other words, a mere paper fix was inadequate. An unflawed computer analysis and associated proper reactor pressure values in Safety Limits 2.1.1.1 and 2.1.1.2 were required to prevent fuel damage. So, why were the affected reactors allowed to continue operating for years until effective solutions that prevent fuel damage were implemented?

2. In its denial of the Technical Specifications Task Force proposal to resolve the problem via a revision to the bases for the Safety Limit, the NRC staff stated that "*from a regulatory standpoint, the proposed change to the TS* [technical specifications] *Bases is not acceptable*." If that proposed fix was unacceptable from a regulatory standpoint, from what standpoint was is acceptable to wait for an average of 8.5 years (see Table 2) to resolve this problem at 28 affected BWRs?

3. Why didn't the license amendment requests submitted after April 2012 (when the BWROG recommended the solution) that sought revisions to the MCPR limits in Safety Limits 2.1.1.2 also revise the directly related reactor pressure values in Safety Limits 2.1.1.1 and 2..1.1.2?

4. How could the NRC approve amendments to Safety Limit 2.1.1.2 after April 2012 that did not also address the associated reactor pressure values in Safety Limits 2.1.1.1. and 2.1.1.2?

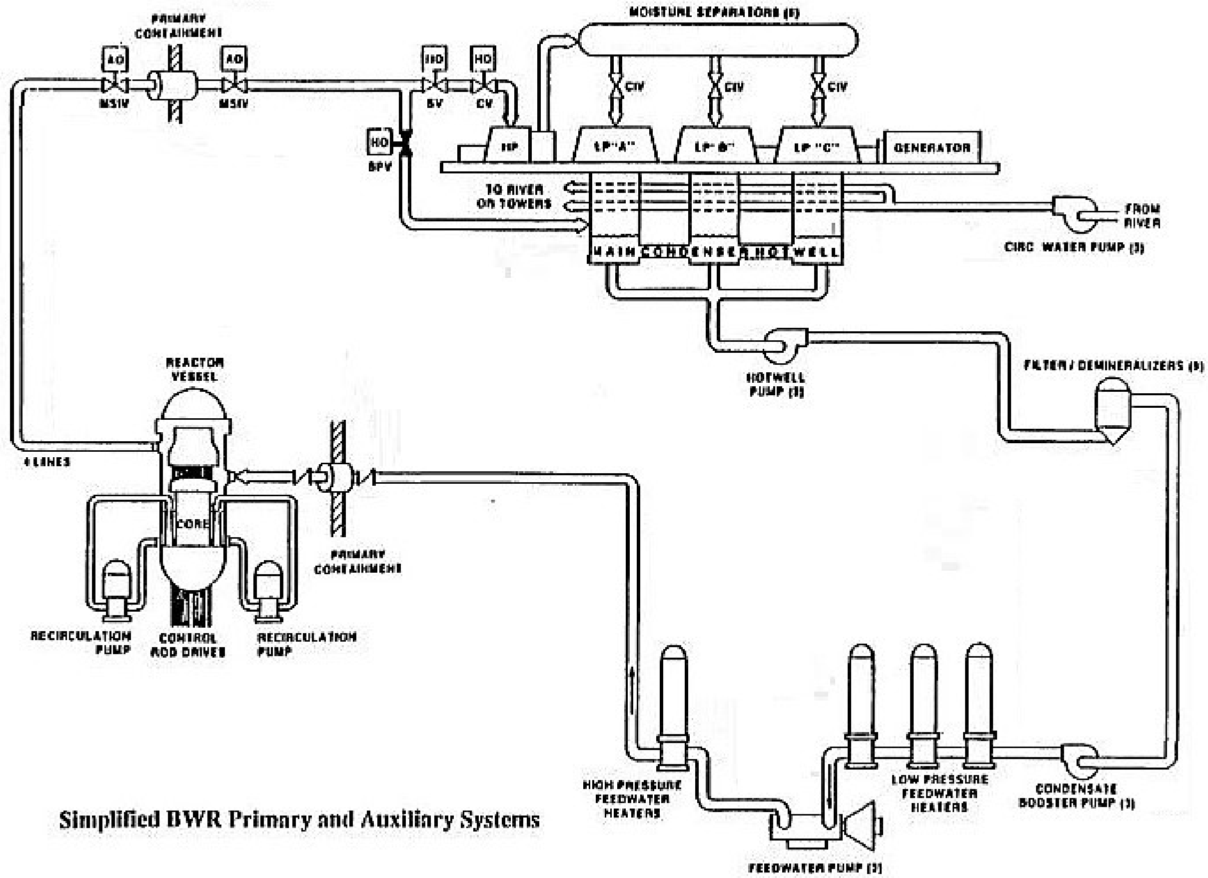
5. Why hasn't the recommended solution been implemented at Brunswick Units 1 and 2, Cooper, Fermi Unit 2, Grand Gulf, and Hope Creek?

6. While many of the amendments that resolved the Safety Limit problem involved just changing the reactor pressure value to match the value used in the safety studies, the solution at Nine Mile

Point Unit 2, and perhaps at other BWRs, also entailed raising the setpoint at which the main steam isolation valves automatically closed to terminate the PRFO event. What assurance does the NRC have that the PRFO event would have been terminated before fuel damage occurred at these BWRs?

7. Had a PRFO event occurred at a BWR between March 2005 when GE reported the problem and when it was resolved and that event resulted in reactor fuel damage, how would the NRC justified to the U.S. Congress and American public allowing that reactor to continue operating despite this known safety hazard?

Figure 1: Boiling Water Reactor Major Systems



Water flowing upward through the core within the reactor vessel is heated to the boiling point. Steam flows through pipes from the reactor vessel to the main turbine. When the turbine is online (as shown in this figure), the steam flows through the fully open stop valves (SV) and the partially open control valve (CV) into the high pressure (HP) turbine. When the reactor is shut down or operating at low power levels with the turbine offline, the stop and control valves are closed. Steam flows through the partially open bypass valves (BPV) into the main condenser.

BWR Safety Limit Problem

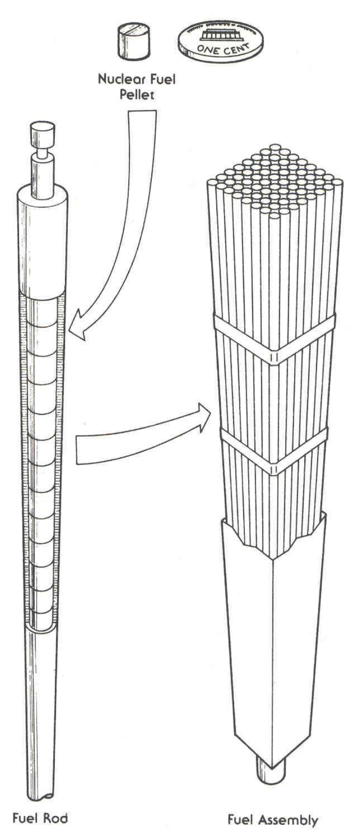
Appendix 1 – BWR Fuel Thermal Limits

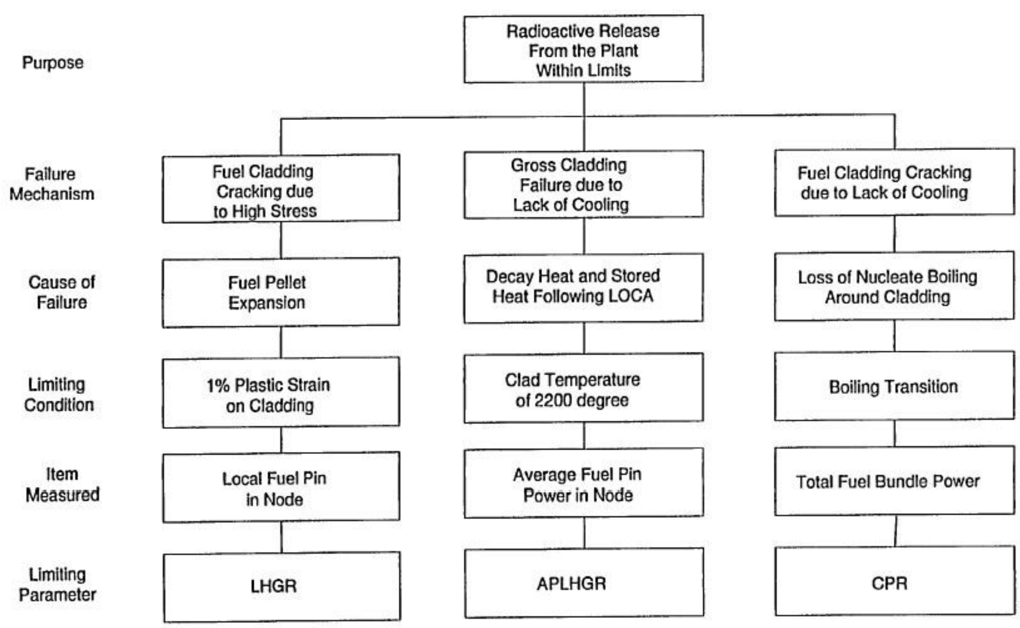
The fuel for boiling water reactors (BWRs) consists of uranium dioxide pellets about the size and shape of a pencil eraser loaded into hollow rods. The rods are long enough to hold 12-feet of stacked pelllets.

The rods are made from zirconium metal and called cladding. Metal pieces, called caps or plugs, are welded onto the top and bottom of each rod to seal it. The sealed fuel rod forms one of the barriers between radioactive material and the environment. While neutons and gamma rays can pass through the cladding, radioactive fission byproducts in the form of particles and gases cannot escape when the cladding remains intact.

Dozens of fuel rods are put together at the factory to form a fuel assembly, also called a fuel bundle. The bottom end of a fuel assembly enables the fuel assembly to be seated in the reactor core and has openings to allow cooling water to flow upward past the fuel rods to carry away the heat produced by the pellets inside. The top of each fuel assembly as a metal handle (not shown in the diagram) that allows workers to move assemblies, one at a time, into and out of the reactor core.

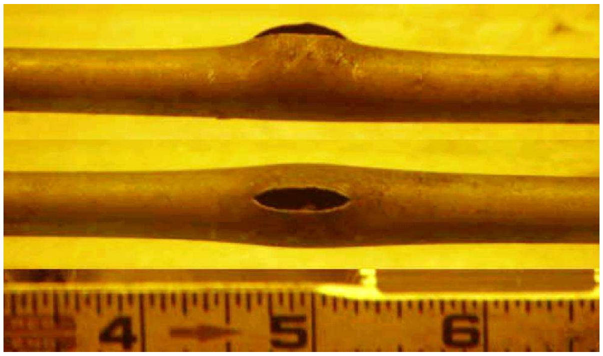
Hundreds of fuel assemblies are placed into the reactor core. BWR cores contain about 80 to 100 tons of fuel, depending on their size.





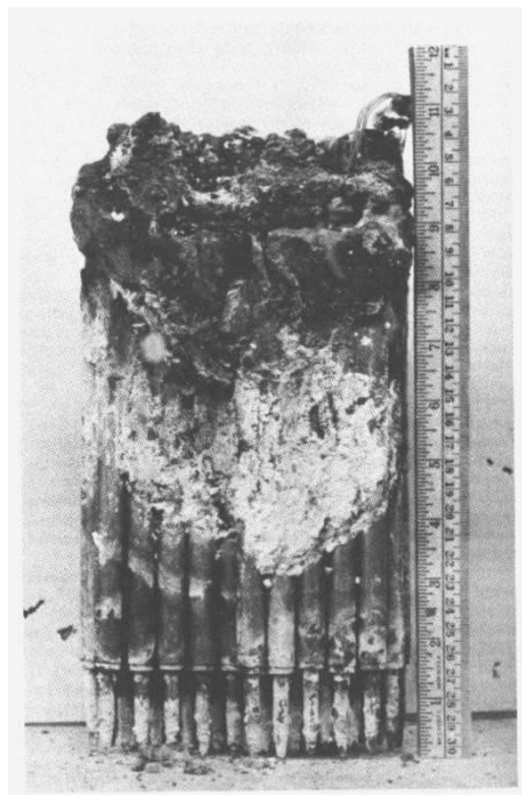
BWR Safety Limit Problem

Boiling water reactors (BWRs) have three fuel thermal limits: the Linear Heat Generation Rate (LHGR), the Average Planar Linear Heat Generation Rate (APLHGR), and the Critical Power Ration (CPR).



LHGR protects against fuel damage during routine reactor operation. A fuel rod is a hollow tube made of metal, called cladding, filled with fuel pellets. As the reactor power level increases, the fuel pellets and cladding heat up. The fuel pellets and the cladding both expand as temperature rises. Because the pellets expand more than the cladding, the pellets put pressure, called stress, on the cladding. LHGR limits the power levels of six-inch long segments of fuel rods to prevent excessive stress from cracking the cladding like the fish-mouth rupture shown here.

APLHGR protects against fuel damage during an accident. If a pipe connected to the reactor vessel were to break, cooling water would jet out from its broken ends. Emergency systems would detect the failure and automatically start to provide makeup cooling water. But in the time needed for these systems to respond—a few seconds—the water level inside the reactor vessel could drop below the core and uncover the fuel rods. APLHGR limits the power level of six-inch long segments of fuel bundles to keep the temperature of the fuel cladding below 2,200°F before the emergency systems can refill the reactor vessel and restore core cooling. The picture to the right shows a fuel bundled that overheated and melted down. The "before" condition of the fuel rods more resembled the section at the bottom than the "blobby" mess towards the top.



EBR-I Core after the meltdown incident.

CPR protects against fuel damage during transients. A transient does not involve a complication like a broken pipe that removes cooling water from the system, but it impedes the ability of the day-to-day systems to handle core cooling. CPR limits the power levels of individual fuel bundles to ensure adequate cooling remains throughout transients. In BWRs, water boils as it flows through the reactor core. This phenomenon is called nucleate boiling because tiny steam bubbles form on the outer surface of the fuel cladding to be swept away by the flow. If bundle power gets too high or flow drops too low, nucleate boiling can be replaced by film boiling. Instead of tiny bubbles (sorry, Don Ho) forming and soon leaving the cladding surface, the bubbles coalesce to form a film along the length of the cladding. This steam blanket impedes the heat transfer from the pellet, through the cladding, to the coolant. As a result, the temperature of the cladding rises to the point where it can blister and crack.