Review of PRA in Regulated Power Reactor Applications

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Abstract

Examples of PRA applications used in regulatory decision-making are summarized and the relative success of each, from the industry perspective and the Nuclear Regulatory Commission (NRC) perspective, is assessed. The bases for the assessment of success in both perspectives are described. It is concluded that the role of the regulator in reviewing industry risk analyses is inconsistent with the role of the owner-operator of a merchant nuclear power plant in certain narrow, but important areas. Of particular interest is the application of PRA in design-related decisions.

Keywords: PRA, Regulatory, Risk, Commercial Nuclear, NRC

Acronyms

1. Introduction & background

The history of regulatory risk activities is adequately described on the NRC website, http://www.nrc.gov/about-nrc/regulatory/risk-informed/history.html, including the several links on that page. As described there, the NRC is "moving toward a risk-informed, performance-based regulatory framework incrementally–in steps." The NRC first clarified its expectations on the use of Probabilistic Risk Assessment (PRA) in 1995 (OIG, 2006). In 1995), the NRC had proposed an agency-wide PRA Implementation Plan that would "provide the necessary interoffice framework for strengthening and increasing the use of PRA technology in agency regulatory activities." (USNRC, 1994). In consideration of the implementation plan, the NRC refers to the Presidential Executive Order 12866 that, along with other guidance, asks regulatory agencies to consider the degree and nature of risks posed in setting their regulatory priorities, as well as costs

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and benefits of intended regulation. A final policy statement was issued that followed closely the proposed rule (NRC, 1995).

In that early time-frame, the NRC envisioned the proposed plan would be beneficial to

> 1. *improved regulatory decision-making,*
> 2. *more efficient use of agency resources in focusing efforts on the most safety-significant issues, and*
> 3. *reduced industry burden in responding to less safety-significant issues.*
>
> USNRC (1994, page 2)

In that proposed plan, the NRC goes on to clarify (in our opinion) an important point of view regarding the use of PRA in the regulatory framework:

> *An important aspect of the expanded use of PRA technology in reactor regulation will be a strengthening of NRC's defense-in-depth philosophy by allowing quantification of the levels of protection and by helping to identify and address weaknesses in the physical and functional barriers, should they exist.*
>
> USNRC (1994, page 3)

In the authors' opinion, there is a tension between items 1 to 3 above and the point of view expressed in the remarks that follow. That is, items 1 to 3 appear to ask for a cost-based focus where (perhaps) excess resources are applied to design and regulation; whereas the remarks that follow appear to seek to strengthen risk-adverse approaches (to the extent they ask for additional focus on design and resource allocation). This tension between finding potential relaxing of overreaching regulatory burden and maintaining or even strengthening of deterministic requirements is at the heart of what became referred to as 'risk-informed' decision making. We believe that stakeholders (the regulator, industry, and the public) should work together to optimize application of risk-informed activities such that the interests of each are best served. This is explored further in the following.

Guidance to industry practitioners for implementing the NRC policy statement was first provided in 1998 and subsequently revised twice over about 13 years to its current form (NRC, 2011). In February, 2011 the NRC commissioned a task force to assess "Options for More Holistic Risk-Informed, Performance-Based Regulatory Approach". The task force had as its objectives to develop a strategic vision and options for adopting a more comprehensive and holistic risk- informed, performance-based regulatory approach for reactors, materials, waste, fuel cycle, and transportation that would continue to ensure the safe and secure use of nuclear material.

The task force was to identify the options and specific actions that the NRC could pursue to achieve a more comprehensive and holistic risk-informed, performance-based regulatory structure and address the following basic questions:

1. Are the current practices adequate for accomplishing the goal of a holistic risk-informed and performance-based regulatory structure?
2. How effective have past and on-going risk-informed initiatives been? What are the relevant lessons learned from these initiatives?
3. Should the use of risk information continue to be voluntary?
4. How effective have recent major deterministic licensing actions (i.e., license renewals, power uprates, B5b mitigation strategies) been? What are the relevant lessons learned from these actions?
5. What are the visions for a holistic risk-informed, performance-based regulatory structure for reactors, materials, waste, fuel cycle, and security?
6. How can the transition from the current system to a more holistic risk-informed, performance based regulatory structure be optimized?
7. What is the schedule for achieving this regulatory structure?
8. How should this structure be implemented?
9. How should stakeholder input be considered?
10. In each area, what are the capabilities and limitations of current probabilistic risk assessment methodologies?

We believe the enumerated questions above help expose fundamental elements of uncertainty the NRC has regarding use of risk in regulation. As external pressures increase, the NRC is faced with meeting its obligation of "adequate protection" realizing that somehow, cost of plant operation should be bounded.[1]

We largely agree with Doorn and Hansson who conclude, regarding PRA and deterministic design (use of safety factors),

> "...the two should be seen as complementary rather than mutually exclusive. Using PRA may lead to a one-sided focus on those dangers that can be assigned meaningful probabilities. Since not all dangers can be quantified and since most decision making is done under conditions of uncertainty, PRA cannot provide the final answer to safety issues. This holds even more when security threats come into play. On the other hand, when optimization becomes important–be it in the prioritization of maintenance measures or in situations where we are faced with hazards that cannot be eliminated–PRA can be an indispensable tool for priority setting and for the effect evaluation of safety measures."
>
> Doorn and Hansson (2011)

We believe that Doorn and Hansson have most clearly framed the underlying basis for "risk-informed decision-making" and in so doing, have exposed why some risk-informed initiatives have been successful while others have not. That

[1] for example 'backfitting', 824 F. 2d 108 - Union of Concerned Scientists v. US Nuclear Regulatory Commission

is, as applications have increased uncertainty (epistemic), there is greater reluctance (paradoxically) to apply risk in decision-making; in applications PRA can be used to evaluate and rank problematic sequences or where comparisons can be made for prioritizing risk reduction measures (Doorn and Hansson, 2011).

2. Commercial reactor applications

In the following Table I, we provide some examples where we believe the PRA has been successfully used and the implications to safety and Nuclear Power Plant (NPP) profit.

Table I: Review of major NRC risk initiatives with comments on outcomes.

Initiative	Description	Implemented
1	Revise TS end-states for required plant shutdowns to avoid cold shutdown conditions	No
2	Revise TS to accommodate a missed surveillance test	No
3	Revise TS to allow startup with inoperable components provided risk is assessed and managed	Yes (3.0.4(b)
4a	Revise TS to increase allowed outage times for individual specifications	Yes
4b	Multiple Specification Risk Informed Completion Times	Yes
5a	Relocate Individual Surveillance Requirement Frequencies	Yes
5b	Relocate SR Frequencies to Licensee Control	Yes
6b	Provide Conditions to Allow Time to Restore from a Loss of Function	No
6c	Provide Specific LCO Time for Conditions Requiring Immediate LCO 3.0.3 Entry	No
7a	Impact of Non TS Design Features on Operability Requirements- Barriers	No
7b	Impact of Non TS Design Features on Operability Requirements - All Other SSC	No
8	Remove or Relocate System LCO That Do Not Meet the 4 Criterion of 10 CFR 50.36 From Technical Specifications	No
Fire	Risk-informed Fire PRA	Yes

3. Industry perspective

The approximate annual cost for maintaining a PRA at a dual-unit commercial nuclear power site, including labor and contracted expenses, is \$1,000,000 yr^{-1}; in today's market, the five year Net Present Value (NPV) on the PRA investment is just under \$5,000,000. When an initiative is undertaken and added to annual expenses (maintenance cost), the five year NPV, ignoring NRC review fees, can be significantly higher than for the baseline costs[2]:

- Fire PRA NPV: \$2,200,000 – \$6,325,000

[2]see for example estimates from the NRC website, ML13004A391

- Seismic PRA NPV: $2,300,000 – $5,200,000

One may wonder why, in the heavily regulated commercial nuclear reactor design and maintenance domain (referring that is, to Title 10, Part 50 of the US Federal Code of Regulations) , would investors and owners feel a need to add the additional costs of PRA? That is, the liability implied by PRA estimates is largely removed by compliance with the regulations demanded. Put another way, compliance with regulations imposed on commercial nuclear reactors ensures very low risk (low frequencies) of accidents the PRA estimates. We assert that, absent adequate offsetting costs, there is little incentive for the owner-investor to bear additional costs for maintaining and developing PRA beyond any existing legally-required minimums.

4. Regulator perspective

The NRC is tasked to ensure "adequate protection" is done regardless of cost considerations[3] consistent with meeting a moral hazard obligation. The adequate protection standard is ensured by regulation using deterministic methods (United States, 1995, III(A)) against the General Design Criteria (GDC)[4] although, in our opinion, the NRC recognizes that, given a hazard such as radioactive release, exposure to harm (or 'inadequate protection') remains regardless the level of safeguards provided (as asserted in Kaplan and Garrick, 1981, for example).

PRA clearly has merit to the regulator faced with a relatively enormous set of scenarios that could have end states of radiation release with possible harmful consequences to the public they are tasked to protect. In its role, the regulator is most interested in finding the many scenarios with harmful end states in order to help identify how it may be possible to avoid realizing them in an application. That is for example, seeing the Large Early Release Frequency (LERF) measure is small is good; but when evaluating a design or design change, it is more interesting from the regulator's perspective to understand the scenarios rather than their frequency.

5. Risk applications

In the following, we explore four risk initiatives; Initiative 4b, Initiative 5, NFPA 805, and Generic Safety Issue 191 - the NRC Generic Safety Issue number 191 (GSI-191) Option 2 (Vietti-Cook, 2010) from the industry perspective and the NRC perspective.

[3]824 F. 2d 108 – Union of Concerned Scientists v. US Nuclear Regulatory Commission
[4]Appendix A to Part 50 – General Design Criteria for Nuclear Power Plants

5.1. Industry perspective on Initiative 4b

Initiative 4b, commonly referred to as Risk Managed Technical Specifications (RMTS), uses "risk estimates" for Incremental Core Damage Probability (ICDP) and Incremental Large Early Release Probability (ILERP) with a fixed Backstop Completion Time (BSCT) to determine the Risk-Informed Completion Time (RICT) (the maximum time allowed before plant shut down) a particular equipment configuration is allowed. The method was formally introduced in 2006 (NEI, 2006) and is implemented since 2007 at one plant site (two reactors) in the US (Yilmaz et al., 2009; Kee et al., 2008).

The plant site using RMTS has implemented it by computing and then storing Core Damage Frequency (CDF) and LERF for each maintenance configuration encountered during power operation. In general each of these maintenance configurations would contain RMTS equipment and non-RMTS equipment (either in the Technical Specifications or not).

Referring to Figure 1, each general maintenance state encountered during a Risk Managed Action Time (RMAT) would have a growth rate (slope) s_1, s_2, s_3, s_n corresponding to either the CDF or LERF stored in the database as well as the length of time d_1, d_2, d_3, d_n after the first RMTS component became inoperable (INOP). The RICT is computed for the most recent (nth) maintenance state based on the corresponding limit, L, for either ICDP or ILERP from the following:

$$\begin{aligned} t_n &= d_{n-1} + \frac{L - h_n}{s_n}; \\ h_n &= h_{n-1} + s_n \left(d_n - d_{n-1} \right); \\ n &\geq 1; \end{aligned}$$

and from the most limiting RMTS component, corresponding to the RMTS component in the maintenance state INOP for the longest time in the most recent maintenance state. The most limiting RMTS component INOP time is 30 days and the RICT can not go beyond the most limiting RMTS component's BSCT.

In summary, RMTS allows more flexibility for compliance with plant shut down or reduced power when in certain Limiting Condition for Operations (LCOs); LCOs occur when certain equipment are out of service and normally, the plant Technical Specifications would require reduced power or plant shut down after fixed period of time. RMTS allows the time to be based on the risk measures mentioned above.

In order to understand the potential impact RMTS may have on plant operational impact, the NRC Licensee Event Report (LER) database was queried for production loss events attributable to Technical Specifications. The resulting records (32) were further reduced for RMTS applicability resulting in 2 reduced to be applicable (Table II).

RMTS has limited application within the plant Technical Specification scope. In particular, Reactor Coolant System (RCS) pressure boundary leakage is disallowed by plant Technical Specifications and is outside the scope of RMTS;

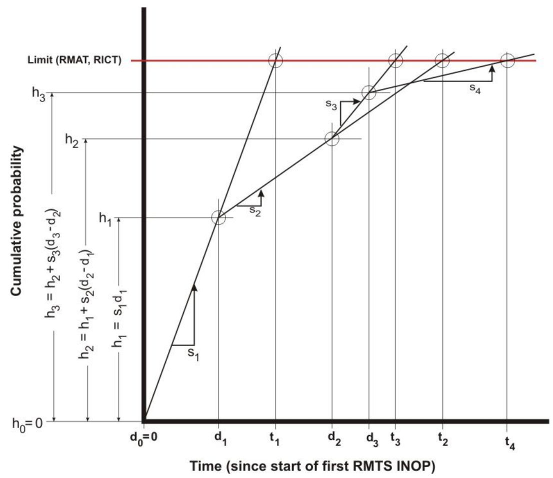

Figure 1: Development of the RICT based on the limit, L, the frequency (CDF, LERF) and the time after entering into an RMAT that each maintenance state ends, $d_1, d_2, ..., d_n$

Table II: US plant shut downs required by Technical Specifications since 2006 showing those avoidable using RMTS

ID (Table III)	Date	Applicability to RMTS
2542009004	09/08/2009	Not Applicable
2632014001	01/17/2014	Not Applicable
2692013004	11/11/2013	Not Applicable
3172012002	07/17/2012	Not Applicable
3342006002	05/26/2006	Applicable
3892014001	07/25/2014	Not Applicable
3892015001	04/11/2015	Applicable
4002013001	05/15/2013	Not Applicable
4832015001	07/23/2015	Not Applicable

the 'Not Applicable' records found since 2006 correspond to pressure boundary leakage (Table II).

5.2. Initiative 5 industry perspective

Under Initiative 5, the NPP operator can use flexible surveillance frequencies based on a risk assessment. The underlying idea is that longer surveillance intervals would reduce unavailability for critical equipment without significantly impacting safety. A motivating factor for industry is reduced maintenance costs (parts and labor) for surveillances. A complicating factor is that is difficult for owner-investors to "see" cost savings unless (primarily) it can be shown that a maintenance or engineering force reduction can be tied to the project. That is, most of the cost for operating and maintaining a NPP is in labor.

Several NPPs have implemented Initiative 5. The annual cost savings for a dual-unit NPP can be as high as $250,000 assuming all avoided labor costs can be recovered. If realized, the savings would likely offset the additional costs associated with maintaining the application.

5.3. NFPA 805 industry perspective

NPP fire protection has been a concern for several years; initial attention brought on by the well-known Brown's Ferry event in 1975. In general, attempts to assess occurrence rates for consequential fires have been consistently high (see Hockenbury et al., 1981, for example) compared to the data (see Gallucci and Hockenbury, 1981, for statistical analysis).

Approximately 50 NPPs have adopted NFPA 805 or are in the process of adopting NFPA 805. As previously mentioned, the cost for implementing the fire PRA can be quite high (more than about $6,000,000) but the business benefit for doing so is unclear to most adopters. The industry consensus is that the costs for implementing NFPA 805 are much higher than expected and the fire risk developed from the analyses is substantially overestimated[5].

There are many reasons for both the costs and risk estimates to be high. Adding the fire analysis as prescribed in regulatory documents (NRC, 2009, is the primary reference) to an already complex PRA is an enormous effort. High risk numbers probably arise due to NPFA 805 asks for relatively high estimates for fire progression extent and initiation frequency (or inappropriate initiator data) resulting in an analysis that is closer to deterministic than probabilistic. Because the cost and lack of perceived safety benefit, we believe the industry feels the NFPA 805 initiative provides little benefit at an unreasonably high cost.

5.4. GSI-191 Option 2b industry perspective

The industry has been working on using PRA to resolve the long-standing NRC safety issue, GSI-191. Because resolution of GSI-191 using a deterministic approach would require large worker dose and high design change costs, the industry has been with the NRC to develop an approach and risk guidance (NRC, 2016), currently drafted and, as of this writing, under review.[6] Because GSI-191 issue in the plant design domain, the issues raised in Section 5.3 are relevant to its resolution. The kinds of issues related to use of high rates of initiation and event progression in the fire PRA are issues in the GSI-191 PRAs, leading to high costs (up to about $7,000,000) and unrealistic occurrence frequencies.

A pilot project was developed that started with a full PRA (Mohaghegh et al., 2013) and, in order to simplify the analysis, is changed to minimize use of PRA and rely primarily on test data (Kee, 2015). After many years of work, no NPP has been able to reach closure with a risk assessment.

[5] for example see ML140619 on the nrc.gov website

[6] letter from Dennis C. Bley, Chairman, ACRS, to Mr. Victor M. McCree, USNRC, 04/19/2016, REGULATORY GUIDE 1.229, "RISK-INFORMED APPROACH FOR ADDRESSING THE EFFECTS OF DEBRIS ON POST-ACCIDENT LONG-TERM CORE COOLING"

5.5. NRC perspective on Initiative 4b

NPP design standards (primarily, Appendix A to Ch. 10 CFR§50) require safety-critical redundant and defense-in-depth systems to be operable at all times; a clearly over-ambitious goal. In order to allow for testing and maintenance (preventive and corrective), Allowed Outage Times (AOTs) were assigned in specific equipment configurations. That is, when certain equipment are out of service, the amount of time is limited as defined in the NPP Technical Specifications (TS). AOTs were developed primarily absent quantitative risk analysis, but rather engineering judgement.

Although the design requires all equipment to be operable, some 'small' amount of out of service time is reasonable. As a consequence, Initiative 4b is only loosely connected to the NPP design domain. The TS also require that any equipment out of service configuration not otherwise specified requires a very short amount of time (usually 1 hour) before the NPP must be shutdown to a safer operating mode. Because of the loose connection to design and the qualitative nature of the allowed out of service times, it makes sense that the regulator would see use of PRA for AOTs.

5.6. NRC perspective on Initiative 5

As in Initiative 4b, Initiative 5 deals with timing, in this case, the time between surveillance intervals (also required in TS. And again, the surveillance intervals are arrived at using a primarily qualitative approach. The regulator has an interest in better understanding of the risk for adopting a specific surveillance frequency for safety-critical systems.

The surveillance frequencies are only loosely tied to design and are specified in the NPP TS. Again, because of the loose tie to design, lack of previous quantitative analyses, and the benefit of a more comprehensive understanding of the interval risk, the regulator is motivated to encourage NPPs to adopt a PRA for surveillance intervals.

5.7. NRC perspective on GSI-191 Option 2b

GSI-191 is in the design domain which has established quantitative methods and evolved standards for equipment performance (primarily, 'test for success'). The design domain relies on quantifiable 'deterministic' methods (see OIG, 2006, for discussion of terms) rather than engineering judgement or the qualitative methods alluded to in Sections 5.5 and 5.6 As a consequence, the regulator is more comfortable with use of design standards in the GSI-191 resolution. The important PRA contribution (in the regulator's mind) is that any scenarios overlooked in the design can be disclosed using PRA.

5.8. NRC perspective on NFPA 805

NFPA 805 again is in the design domain. Fire standards have been accepted and used for many years building and process industry applications. Just as in GSI-191, the regulator is less interested in the LERF and more interested in scenarios that could have been missed when designing the plant. Here again, the design standards that use 'test for success' are the basis for risk assessments.

6. Conclusions

We have briefly reviewed some of the risk-informed activities that are being either used, or contemplated to be used, by the NRC and the US nuclear fleet. Based on a review of the underlying work by the NRC, and our experience in industrial application of some of the activities, we believe there exists a tension, rooted in perspectives of intended use between the NRC objectives and industrial application of risk-informed activities.

Industry sees opportunity in risk-informed activities to improve cost performance while maintaining or improving upon high levels of safety; the NRC primarily sees opportunities to improve on the existing regulatory structure with less emphasis on cost performance; both perspectives consistent with the respective organization's goals. A third stakeholder, seemingly less engaged in evaluating application of risk-informed activities is the public, a consumer of the both the industrial production and regulator oversight. The public has interest in both safety and cost (of product consumed).

In the authors' opinion, as risk-informed activities go forward, it may be beneficial to stakeholders (industry, the regulator, and the public) to explore new ways, or build upon existing approaches, whereby the all perspectives could be effected to optimize performance among all stakeholder interests.

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7. Event Reports

A search for LERs since 2006 on the NRC website using the keywords {Technical Specifications AND shut down} produced 32 results (as of 26 April, 2016).

Table III: Failure data from the USNRC LER database with date (Date), cumulative count (Cum), trip or other transient (t/o), and narrative description (Narrative)

LER Number	Event Date	Abstract
2542009004	09/08/2009	**Pinhole Leak in Core Spray Piping Results in Loss of Containment Integrity and Plant Shutdown for Repairs** On September 8, 2009, during the performance of a quarterly 1A Core Spray (CS) system [BM] flow test, a pinhole leak was identified in the 1B CS minimum flow line just downstream of MO 1-1402-38B (1B CS minimum flow valve [V]). The leaking pipe [PSP] could not be isolated from primary containment [NH]. As a result, primary containment was declared inoperable at 1935 hrs and Technical Specification (TS) 3.6.1.1, Required Action B.1, was entered requiring a plant shutdown in 12 hours. Unit 1 was shut down on September 9, 2009 at 0638 hrs. This event is therefore reportable under 10 CFR 50.73(a)(2)(i)(A), "The completion of any nuclear plant shutdown required by the plant's Technical Specifications." After Unit 1 was shutdown the affected CS piping (5 inches long by 1-1/2 inches in diameter) was removed and an inspection was performed. The inspection confirmed that an area of the pipe had severely thinned. The thinned area had indications consistent with cavitation erosion. The 1B CS piping downstream of 1-1402-38B was replaced on September 10, 2009, and a section of the 1B CS piping upstream of 1-1402-38B was also replaced on September 10, 2009. Unit 1 was synchronized to the grid at 0822 hrs on September 12, 2009
2632014001	01/17/2014	**Primary System Leakage Found in Recirculation Pump Upper Seal Heat Exchanger** On January 17, 2014, leakage into the Reactor Building Closed Cooling Water (RBCCW) System was determined to be Reactor Coolant Pressure Boundary (RCPB) leakage as identified by the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS). Based on this, the TS limiting condition for operation was not met and a plant shutdown was required. The plant shutdown commenced at 2029 on January 17, 2014. There was no radioactive release from the plant. The plant was shut down without incident to repair the source of the inleakage. The apparent cause for the RCPB leak was the lack of an established maintenance strategy in place to periodically check the condition of the heat exchanger or replace it. A crack formed in the #12 Recirculation Pump Upper Seal Heat Exchanger coil due to intergranular stress corrosion cracking. The leaking #12 Recirculation Pump Upper Seal Heat Exchanger was removed and the system was modified to operate without this heat exchanger by utilizing the excess capacity of the #12 Recirculation Pump Lower Seal Heat Exchanger.
2692013004	11/11/2013	**High Cycle Fatigue Resulted in Reactor Coolant Leak and Unit Shutdown** On November 8, 2013, the Oconee Unit 1 Control Room received an alarm associated with the containment atmosphere particulate radiation monitor. Reactor Coolant System (RCS) leakage of <0.1 gpm was identified. On November 11, 2013, upon verification of un-isolable reactor coolant system pressure boundary leakage on the 1B2 High Pressure Injection (HPI) Injection line, Oconee Unit 1 was shut down as required by Technical Specifications. The shutdown was orderly and without complication. The cause evaluation determined that mechanical, high-cycle fatigue resulted in a through wall crack in the stainless steel butt weld between the HPI nozzle safe end and HPI piping. Ownership and oversight of the augmented examination program was inadequate, as was guidance to the examiners for actions to be taken when full weld volume coverage could not be achieved. This event is reportable under 10 CFR 50.73(a)(2)(i)(A), as completion of a shutdown required by Technical Specifications, 10 CFR 50.73(a)(2)(i)(B), operation or condition prohibited by Technical Specifications, and 10 CFR 50.73(a)(2)(ii)(A), degradation of a principal safety barrier. High pressure injection capability was maintained, and containment integrity was not impacted. This report has been revised to reflect the results of the completed root cause evaluation.

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LER Number	Event Date	Abstract
3172012002	07/17/2012	**Reactor Coolant Pressure Boundary Leakage due to Tubing High Cyclic Fatigue** On July 17, 2012, Reactor Coolant System pressure boundary leakage was determined to exist on Unit 1 11A Reactor Coolant Pump differential pressure transmitter tubing. Operators commenced a Technical Specification required unit shutdown. With reactor power at 10 percent a containment entry was made to isolate the leak. This effort stopped the steam emanating from the insulated tubing. Unit 1 returned to full power. Unit 1 leak rate data was monitored for the next several days. It was determined conditions did not improve as expected. An additional containment entry was made on July 21, 2012 which identified that Reactor Coolant System pressure boundary leakage existed past the previously shut isolation valves. Operators conducted a Technical Specifications required shutdown of Unit 1 to MODE 5. The source of the leak was a crack in the tubing side weld of the pipe to tube adapter. The cause of the leak was high cyclic fatigue. The cyclic fatigue was caused due to a vertical support for the tubing that was not connected. Corrective actions included replacement of the adapter, the affected portion of tubing, and the connection of a re-engineered vertical support. The similar welds on the other Unit 1 reactor coolant pump differential pressure transmitter tubing runs were inspected with no issues identified. Unit 1 returned to full power on July 25, 2012
3342006002	05/26/2006	**Unit Shutdown Completed as Required by Plant Technical Specification for Failed Solid State Protection System Memory Card** At 1046 hours on May 26, 2006, Beaver Valley Power Station (BVPS) Unit No. 1 removed Train B of Solid State Protection System (SSPS) from service to perform a routine bi-monthly channel functional surveillance test in accordance with Technical Specifications. The Unit is required to enter multiple BVPS Unit No. 1 Technical Specification Action statements for the SSPS being out of service. At 1145 hours, the Memory test portion resulted in failed indication from four memory test positions. At 1415 hours, BVPS Unit No. 1 commenced power reduction to comply with the most limiting Technical Specification Action since the memory tests were still failed and Train B remained inoperable. At 1831 hours, the reactor was shut down per the normal shutdown procedure and Mode 3 (Hot Standby) was entered. This event is reportable pursuant to 10 CFR 50.73(a)(2)(i)(A) for completing a plant shutdown required by the plant's Technical Specifications. The root cause was indeterminate despite rigorous investigation. The most probable cause of the event was a poor or intermittent electrical connection in the testing circuit of Solid State Protection System. The suspect connection was narrowed down to one of three locations: Pin 4 of Universal Card A412 or the rack connector or termi-point clip for this pin. This failure would not have prevented the proper operation of the SSPS since Pin 4 is used only during testing and not during normal operation of the SSPS train. The failure of the SSPS Train B test circuit was very low risk significance.
3892014001	07/25/2014	**Unit Shutdown due to Leak on Safety Injection Tank Vent Valve Piping** On July 25, 2014 with St. Lucie Unit 2 in Mode 1 at 100% power, a leak was confirmed on a one inch pipe between a safety injection tank (SIT) and a discharge header vent valve. In accordance with Technical Specifications (TS) and plant procedures, operators subsequently shut down the unit to repair the leak. The shutdown was uncomplicated and all plant safety systems functioned as designed. The leaking vent line and valve assembly were replaced and returned to service on July 28, 2014. Engineering evaluation identified the direct cause of the pipe leak as through-wall cracking from high cycle, low stress fatigue. This condition is reportable in accordance with the following requirements: 1) 10 CFR 50.73(a)(2)(ii)(A), 2) 10 CFR 50.73(a)(2)(i)A, 3) 10 CFR 50.73(a)(2)(i)B, 4) 10 CFR 50.73(a)(2)(v)(D), 5) 10CFR50.73(a)(2)(ii)(B) and 6) 10 CFR 50.73(a)(2)(vii)(B). This supplement revises the event description, analysis of event and safety significance and adds additional reporting criteria. This condition was determined not to be a significant impact on the health and safety of the public.
3892015001	04/11/2015	**Unit 2 Shutdown due to Through Wall Crack and Leak in the 2B2 Safety Injection Tank Discharge Pipe** Abstract: On 4/11/2015 at 1204 EDT, a through wall leak was identified during an investigative walk down of the 12-inch diameter Class 2 piping for the 2B2 safety injection tank (SIT) discharge header. Operators declared the SIT inoperable and Technical Specification Limiting Condition of Operation (LCO) 3.5.1 action "b" was entered, which required the SIT to be restored to Operable status within 24 hours or shut down to Mode 3 within the next 6 hours. In accordance with Technical Specifications (TS) and plant procedures, operators subsequently shut down the unit to repair the leak. The shutdown was uncomplicated and all plant safety systems functioned as designed. The leaking piping was replaced and returned to service on 4/18/2015. A sample of the failed pipe was sent to a metallurgical laboratory for examination. Results of optical and electron microscopic evaluations revealed the cause of failure as low stress high cycle fatigue. This condition is reportable in accordance with the following requirements: 1) 10 CFR 50.73(a)(2)(i)A, "The completion of a nuclear plant shutdown required by the plant's Technical Specifications," and 2) 10 CFR 50.73(a)(2)(i)(B), "Any operation or condition which was prohibited by the plant's Technical Specifications." This condition was determined not to have a significant impact on the health and safety of the public, since the identified leakage was well within the plant capability to maintain SIT requirements and was insignificant as compared to either the SIT discharge flow rate or the safety injection flow rate during design basis events.

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LER Number	Event Date	Abstract
4002013001	05/15/2013	**Reactor Pressure Vessel Head Penetration Nozzle Indications Attributed to Primary Water Stress Corrosion Cracking** Abstract: On May 15, 2013, while at 98% power in Mode 1, HNP commenced a Technical Specification required shut down to Mode 6 to repair a flaw that was identified in nozzle 49 of the reactor pressure vessel head. Nozzle 49 was subsequently repaired on May 31, 2013, utilizing the inside diameter temper bead welding process. In 2012, four nozzles (5, 17, 38, 63) were identified with similar indications that exhibited characteristics of PWSCC, and were subsequently repaired using the inner diameter temper bead welding process. However, nozzle 49 was not identified as having an indication at that time. Because the indication in nozzle 49 was identified while at power, a shut down was required by Technical Specifications. The cause of the flaws in nozzle 49 and the other four nozzles was attributed to PWSSC. The root cause evaluation determined that the missed identification of the indication in nozzle 49 was due to the lack of mitigating programmatic governances to specify process independence and fatigue/distraction controls. The planned corrective action to prevent recurrence is to create mitigating programmatic governance for providing oversight for complex automated Non-Destructive Examination (NDE) inspections through the generation of new procedure(s).
4832015001	07/23/2015	**Completion of a Shutdown Required by the Technical Specifications - TS 3.4.13** On July 23, 2015, plant operators became aware of indications of an increase in the Reactor Coolant System (RCS) unidentified leak rate. The indications included containment radiation alarms as well as increasing containment humidity and sump levels. An RCS inventory balance indicated an unidentified leak rate of 1.2 gpm leak which is greater than the Technical Specification limit of 1 gpm for unidentified leakage. Actions were taken to determine the source of the leak. A containment entry was made, and a steam cloud was identified to be coming from the Pressurizer Spray Valve cubicle. The plant was shut down in order to comply with requirements of the Technical Specifications. It was determined that the leak was due to seat leakage through the RCS Pressurizer CVCS Auxiliary Spray Supply Drain valve BBV0400 and then through the non-safety related pipe flange immediately downstream of the valve. The valve was tightened which reduced the leakage to 60 drops per minute. The flange gasket was replaced. Additional causes and corrective actions are still being determined