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Comment On: NRC-2019-0062-0012
Preliminary Proposed Rule Language: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors

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Submitter Information

Name: James August
Address:
ARVADA, CO, 80004
Email: jkaugust100000@gmail.com
Phone: 17068342997

General Comment

Please see the attached files in Word, for convenience, and PDF, to lock. They are identically the same file; Word is provided for your convenience. Since many of these technical concepts on 1980s era operations and 1970s vintage Advanced Reactors are dated, feel free to contact me if you have specific questions about events or other items that I reference. JK August. PE jkaugust100000@gmail.com smart cell 706-834-2997

Attachments

Combined Materials NRC Part 53 NOPR 2022August 31 JKAugust

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August 31, 2022

Notification of Public Rulemaking, (NOPR), Part 53
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Re: Docket ID NRC-2019-0062 –
Public Comments, from JK August

To whom it may concern:

I offer the following suggestions for Part 53. Background details are attached. I don't make a point-by-point review of the proposed rule, as it stands today. I support both Framework A & B approaches, but suggest considering a third alternative, which starts from scratch. I also understand no one wants five different licenses on one plant design*. (Note: Please see the acronym list at the end of Appendix 1 pgs. 16-18 if you encounter an undefined acronym.)*

My suggestions summarized:

1. Give the licensee maximum latitude within a set of top-level constraints. Within any framework, give the designer the options to make alternative proposals.[1] Within one design, use one standard licensing approach.[2]
2. Following traditional conceptual design, alternative risk-informed processes can evaluate safety design risks effectively. Methods besides Probabilistic Risk Assessment (PRA) proposed to evaluate safety risk are encouraging.[3] Others use equally effective alternatives to PRA, or that, exclusive use of PRA is exactly counter to the charge developing Part 53 (Appendix B)
3. Performance Basing (PB): Usually traditional license-required programs under Parts 50 and 52 can be readily performance-based (PB). This includes those like Licensing, Quality Assurance and Maintenance, as examples. Licensing has many legacy commitments that can readily be reviewed for obsolescence based on risk. Utilities don't though, because requesting commitment relief is laborious, time-consuming and frankly, one of the dullest things anyone can do. The last plant I was at had several hundred maintenance commitments stemming from 1980s event. Upwards of a hundred were obsolete, and no longer relevant. They had at least one recent low-level finding for not doing exactly one simple commitment. Risk-wise, the commitment tasks made would have cost millions to perform, while introducing substantial new risk. Nonetheless, their Licensing Group was unwilling to support relief request, so they delegated the task to Engineering to somehow comply – with virtually no resources to do it. This is all too typical of problems associated with licensing commitments. Resolving them would have huge safety benefits. A second example is QA….. Consider the use of Condition Reports to demonstrate regulatory compliance to report every little

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[1] Fort St. Vrain HTGR (FSV) was licensed under Part 50. Post-TMI licensee ineptitude operating the plant introduced numerous political complications. Not all can be attributed to the licensee, PSCo, alone. Operating a nuclear plant presents any new licensee a learning process need; PSCo had an industry of peers learning, too.
[2] Use one licensing method per design.
[3] Writing ANS 53.1 in 2007-2012, my conclusion was that PRA was the exclusive risk evaluation approach.

condition possibly adverse to quality. A typical pant gets maybe 50 per day per unit. That's 100 per day for a two-unit plant, or upwards of 50,000 a year! These numbers are indeed staggering. However, they lend themselves to rapid numerically rendering down. Statistically, they tell a story. Similar complaints and problems crop up, over and over. Unresolved issues that persist over time tell a story. Rarely have I seen that indicator used effectively at any nuclear plant to zero in on unresolved problems. Intense use of this indicator statistically tells another PB story, but again, utilities are not in the habit of statistically evaluating these. A third example is maintenance. Virtually every nuclear plant I've worked at has substantial amounts of uncompleted scheduled PM work orders. Many have been deferred many times. Most have processes for deferrals, and they defer the same materials over and over again, every outage review. Many PM intervals are elastic, that is, they can stretch[4]; about 10% or so statistically, can not. EQ PMs are one example. In fact, utilities only complete a small fraction of the PM work they have on their CMMS programs. It varies, but typically is well under 50%. Some run as low as 15%.[5] A performance-based measure I've proposed is to measure the compliance percentage of PMs on the CMMS scheduled "books". If compliance percentage is low, it means one of two things – the PMs aren't exact ("actionable") and aren't needed, or they're putting important equipment – like EQ qualified lifetime equipment – at risk. This is another example of PB the utility could do on their own that would greatly simplify their work and lower costs, but they won't.[6]

4. Maintenance & Reliability Assurance Programs (RAP): While the current framework provides a D-RAP for design (RAP) and assigns current license practices to cover O-RAP, e.g., Maintenance Rule 50.65, EQ Rule 50.49, other guidance (50.34 Preliminary Design, 50.46, Generic Letter 83-28, Vendor Technical Information use and maintenance….etc.) there has never been specific guidance or request for plants to pre-develop their maintenance plan based on Engineering best practices, methods or standards.[7] Such information is well-developed at this time; the methods are available to do that work better than ever in the past. Without prescribing how, any rule should require that a licensed plant predevelop their complete estimated[8] maintenance plan before the onset of operations. No plant in the US, indeed probably the world, has ever had such a requirement. The primary benefit of having such a plan is to "hit the ground running" by not having to develop maintenance activities and equipment on the fly, in crisis, after operations begin. Such plans would include addressing plans for servicing highly-irradiated equipment before it becomes "unserviceable" by being highly irradiated, with few or ineffective plans on how to manage it. This was the case for over 60 years beginning not only at Fort St Vrain (FSV), but BWRs and PWRs that later had to grapple with how to manage such tasks. Not having such plans with their bases supports continued ad hoc catch as catch can maintenance that drives high cost.

[4] Rarely are general inspections or instrument calibrations for drift taken out to appropriate intervals.

[5] The last plant I was at had around 18,000 PMs for 2 units, 12,000 active. Every outage around 1000 PMs were tasked to systems engineering to decide whether to do or not. Literally hundreds of hours of engineering time went into the same reviews every outage, for which few if any had any specific reliability training to assess – most of whom were direct out of college. This represented a huge opportunity for the utility, yet one they could not address and take advantage of. Engineering turnover, no surprise, ran well over 20% per year.

[6] The implications here are obvious. Substantial amounts of utility scheduled maintenance have little if any benefit as scheduled. A PB approach would be reschedule, replan, or somehow figure out how to get absolutely indisputably necessary work done – like EQ PMs.

[7] This will be contested; some will argue later plants had technical specifications requiring it. They did not.

[8] Guesstimate, most probable, expected level of replacements, inspections, tests and the methods to do them.

5. Performance-based bias: Accept scientific and engineering facts and their bases. Avoid endorsing politically-based, biased third-party reports and interpretations unsupportable by factual data. I allude to issues that set the world against high-temperature graphite-moderated reactors HTGR in the 1980's and 90's post Chernobyl. The controversy over burning graphite, unscientifically based, courtesy of Soviet bloc investigators was unsupportable by physical chemistry published data. Nevertheless, NRC put a hold on FSV startup.[9] Neither reflects well on use of Engineering or Scientific practices. Jointly, these issues and many others refocused substantial efforts away from engineering where they were needed correcting real issues onto irrelevant, inconsequential low-risk distractions. Those impacts were substantial contributions to ending the pilot plant's life. I would like to present myself as a PB expert, but I'm not. I am a keen observer of processes though, with enough experience to think them through. Many utility staff are not, nor do they seek to risk challenging practices as they exist. PB-ing regulations could stimulate this needed improvement.
6. Risk-informing work: the single biggest opportunity I see to risk informing work is to establish a clear guide to what a critical risk is. This involves two aspects of work, the plants risk-based/informed design, including redundancies, etc. and the plants partition down to critical parts. This in turn requires an effective use of the term critical. Critical must be exactly defined; vaguely having the potential to affect a safety system has no potential to make an item critical. To be critical, something must have the certainty, in its failure, to directly fail the critical item one level up. While this sounds like the single failure criterion, it's not. To not explain further, see ASME-RAM-2 (2016). With a defined criticality and a plant partition hierarchy based upon safety role from the critical system design level down to the SSC part and its failure modes and causes, risk informing all work is surprisingly easy. The key requirement is to specify critical and the design basis of the plant, including all redundant safety systems that are expected to be in service. "Direct failure" then becomes the key criteria for ranking all work activity. Risk ranking to support this is rarely found in plant documentation of Generation ½ nuclear plants – those built in the 1970s-1990s. Engineering must develop it. I have spent a career risk ranking nuclear equipment in the past. This broad activity extends well into process support activities. Properly understanding risk allows the parking of substantial amounts of work that needs only to be performed when required, based on evidence of other equipment or systems failures. Doing this well takes the randomness and uncertainty out of performing tasks. SDOs would do well to provide a standard for risking ranking, on this basis.
7. Utility performance: Placing more accountability in the hands of design developers and users increases risk to the NRC. Utilities can be fundamentally not qualified to make some decisions by their governance practice or simple professionalism.[10] An example is authorizing unqualified managers and supervisors to make technical decisions in the face of those already made by qualified designers and engineers. This has been a practice in crises throughout my commercial career. It flies in the face of PE certifications, technical degrees and simple common sense – technical expertise which engineers provide – in principle. Professionally, engineers must disqualify themselves from performing work on tasks they are not qualified to do. When utilities do stupid things, which they do on occasion, they need to be taken to task. In my career I've never seen any organization except the NRC do so. By stupid things, I mean things such as issuing expired materials

[9] In the light of Chernobyl RMBK dismemberment, another would be alleged FSV PCRV construction weaknesses. Ignoring PCRV pressure vessel load redundancy, the facts did not support these issues in any way. Nonetheless they contributed to shutdown orders, raised costs, and encouraged project abandonment.
[10] This issue is not specific with regards to PSCo/FSV; it pertained most to my last direct employee engagement.

from storerooms, or rescheduling “hard time”[11] EQ work beyond scheduled work dates, demanding extensions or that engineers fabricate calculations unsupportable by scientific plant basis, calculation or facts. Usually, these activities are minor in risk, but occasionally they’re not. Over time they add up. Organizations that can’t live by engineering processes, shaving margins, do not deserve their license; they smack of the recent 2018-2021 Boeing 737 Max certification fiasco. This was fraught with MCAS[12] design inconsistencies and failures. (Note that after the fact, suppression of recommendations passed up – by managers – came to the public’s attention for review.) Such practices must be censured at the management level, if indeed they come to pass ever. NRC alone can awaken management, and must be prepared to do so, as it has in the past.[13] It must continue to do so whatever licensing approaches are used. Simple rules and regulations make them easier to understand and enforce.[14] Simpler, more accountable rulemaking should become the complementary factual side of process, management shortcomings.

8. ALARA: As Low as Reasonably Achievable needs to be put onto more common playing field. It has driven up cost and substantially raised risks by enticing Owner operators to put off maintenance on irradiated equipment that could have been done more simply before that equipment failed. AT FSV, this included Control Rod Drive Assemblies, Circulators (RCP Pump equivalents - compressors), refueling equipment, and other equipment. Having HP (or RP, the modern name) decide who will do what maintenance when, holding the trump cards, is not good operating practice. As a maintenance professional, I find begging RP (non-engineering, non-maintenance) staff to do control rod drive PMs preventive maintenance specified totally unacceptable. This has been the case at every nuclear plant I’ve worked at. I can provide the list, and its close to 10. If maintenance will not be allowed above certain doses, that needs to be in the license. Period.
9. Institute of Nuclear Power Operations (INPO): NRC delegates much responsibility to INPO. Thus, they should be subject to formal review. INPO has sponsored ineffective unsound engineering practices unchallenged;[15] they are not an engineering organization. How INPO fits into the license picture needs separate review consideration. Grandfathering all INPO rules and requirements should not be assumed for new licensees. Ideally, safe plant designs would limit requirements like INPO.

Respectfully,
JK August, PE
706-834-2997
Jkaugust100000@gmail.com

Attachments

Appendix 1: Technical Supporting Points

[11] Unextendible; e.g., must be done before the end of its life. Lifetimes are a vague notion to some unqualified staff. That’s why nuclear plants have qualified engineers, buttressed by rules.
[12] Maneuvering Characteristics Augmentation System (MCAS) designed and certified for the 737 MAX to enhance the pitch stability. Ultimately certifying this system Boeing overruled numerous problems and recommendations by pilots & engineers identified during the MCAS development process.
[13] One reason I take most NRC excesses in stride is because the utility operators in question brought them on.
[14] As simple as it sounds, I have explained the concept of expired material lifetimes over many times to licensed nuclear operators and managers, to no avail. There is a point where people must simply follow the rules.
[15] One was/is their arbitrary PM tasks (PMs) extension allowance in their maintenance documents. Life limiting PMs can’t arbitrarily be extended, as suggested in their materials.

Appendix 1: Technical Supporting Points

New Advanced Reactor (AR) Technology Reactor Licensing Issues for thought

Background:

My nuclear involvement has spanned 48 years in multiple reactor technologies. Since 1974, this has included naval PWRs, the Fort St. Vrain high-temperature gas reactor (HTGR), 1974-1995, GE Mark 4 and advanced BWRs, and Westinghouse and CE PWRs. I've performed shorter duration nuclear, fossil utility and transmission distribution reliability projects as well.[16] Much work has centered on developing solutions to equipment problems affecting operations and maintenance. Other efforts focused primarily on engineering administrative support process consolidation streamlining systems to be more effective. This has been a substantial utility need across the board, over my career.

Ironically, one lesson learned was that difficult nuclear engineering (re)design issues during plant operations could have been easily solvable, prescribing correct maintenance beforehand to follow.[17] Engineering redesign in lieu of maintenance is many times (~100) as expensive a way to address a problem.[18] Effective maintenance planning also substantially reduces worker exposure dose. Operational headaches, workarounds, outage cries and other negative impacts are substantially reduced. At any rate, real events from my experience support these comments. Please contact me if you have any questions.

Additional Context:

Concerns that I remember as being especially impacting follow below. These cover emerging advanced reactor technologies like General Atomics HTGRs, as well as mature LWR technology. In both LWR and advanced reactor situations, the immaturity of early nuclear generation design and operations in the 1960s, 1970s and 1980s led to issues that in retrospect were predictable. Had they been considered earlier in design phase, a significant number of these issues could have been more effectively addressed.[19] This applies especially to maintenance of highly irradiated in-core nuclear equipment like control rod absorbers. To develop NOPR comments supporting Part 53, some defining events are provided. I subcategorize issues as (1) ones that affect all new nuclear plants, (2) issues faced unique to new non-LWRs AR designs that ideally technology neutral licensing would further address and manage to develop consistently[20], and (3) other issues that all plants face, regardless of technology, engineering philosophy maturity and specific operations approach. Fort St. Vrain (FSV) had a Commercial License "rise-to-power" license. Each step of the license (stepwise power rise to 100% full 860 MWth rated thermal power) depended on successfully completing the previous steps. Once problems arose, step completion success was contentious or impossible. As one result, the plant never operated

[16] One to six years.

[17] That is, an effective maintenance program plans would have enabled more adequate maintenance performance on items that were later submitted for redesign design changes (DCs) to address. This lesson applies to all reactors, but especially to first of a kind reactors like FSV.

[18] I base this statement on initiation of design changes and eventual proposed DC review of costs.

[19] Many were not and could not be addressed, because the design wouldn't allow it. Designers had presumed a dry Helium environment for many parts that eventually failed due to cracks from fatigue and SCC cracking.

[20] Address fairly, based on that technology's licensed risk, not another technology's risk -- like LWRs. Fort St. Vrain (FSV) routinely ran the LWR litmus test, inappropriately, in my opinion. I cannot list enough instances here.

commercially at full original design rated power for any length of time.[21] Instead, holds at 30%, 40%, 60% and 84% limited it to a fraction of the original design's 860 MWth rated thermal power at various points from 1976 to 1988 shutdown. For a brief two-week period in late October 1981, the plant operated at full-rated thermal power of 860 MWth with valve wide open (VWO)[22] design generation of 330 MWe net. Many operators claimed it was the smoothest operating period of the plant.[23] While that may be a stretch, there were few unique full power operating issues then that I remember.[24]

1. **All new nuclear plants**
    a. Licensing Hierarchy (safety risk progression down to materials and their specific special treatment (ST) requirements). Specifically, many plants wound up with huge safety systems that only had one or two safety functions that required SSC designated SR. Lack of technical expertise and SSC designer categorization classification meant many bought all SSC within those systems to the highest certified safety treatment standards. NRC inspectors routinely challenged any use of NSR components bought commercially without certifications to only commercial grade standards for nonsafety related SSC. Many components and systems at FSV fit this mold, greatly increasing costs of operation with custom replacement parts. FHM and BUBW systems at FSV were notable in this regard.
    b. First plant prototype application – design demonstration requirements (rise-to-power license with NRC power rating holds). Ideally no commercial power plants would be licensed in this manner; they would be proven as viable commercial technologies first.
    c. Quality (big "Q") in relation to Plant Reliability and Availability[25]: Emphasis needs to shift towards performance-based quality requirements. The single biggest improvements would come in (1) complete delineation of SSC types, SR, NSR and NSRWST (or other category) based on how the design license does this.[26] Then (2) SSC special treatment requirements must be utterly clear, enough to specify exact, defensible STs for covered equipment. The SSC equipment partition must be clear, from top-level safety function to lower most SSC material physical degradation failure requirement. Vague inexact non-specific (hence non-actionable ST) should not be allowed to develop later, without specific conditions for doing so. It is understood, a graded approach to QA on in-scope SSC is taken.
    d. Process quality emphasis would introduce far more potential for performance-based improvements. Commercial U.S. nuclear plants fought non "big-Q" quality initiatives tooth and nail. Unfortunately, they face extremely high costs as one result due to heavy rework

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[21] For a ten-day period in 1981, it operated at 100% rated thermal power.
[22] Valves wide open (VWO): the way plants normally operate at full steam load to the turbine. (Turbine admission valves wide open)
[23] NRC gave special permission exemption to restrictions for this full-power run. A "Loop Shutdown" RPS event for a false positive noise terminated it.
[24] Technical literature naysayer stories in the popular press say the plant never operated at full rated load, which is not the case.
[25] Quality (Big "Q" and quality (little "q") I use to refer to prescriptive deterministic quality like Part 50 Appendix B and process statistic quality as reflected by ISO 9000-type standards. I see more room to introduce performance based little q quality into nuclear design and operations processes. Part 53 should consider this.
[26] This is a complex subject, but better methods of partitioning SR SSC parts in SR and NSR subassembly parts is imperative to apply effective ST where they are appropriate

loads in all areas.[27] Process quality initiatives could identify and address those many shortfalls. Those "little q" quality program tools famous from Deming and Japanese use (statistical analysis of "failures", failure analysis such as Hosakawa "fishbone" diagrams, root cause analysis and proactive quality initiatives would benefit nuclear processes that are ingrained, ineffective based on statistical analysis and haven't fundamentally improved for years, witness high nuclear operating costs. While I don't believe in proscribing those specifically, I do believe that requiring demonstrated failure statistical improvements would logically move licensees toward better methods proven elsewhere. Including an annual review of licensee improvements performance might initiate efforts in that direction.

e. Graded approaches to Quality: as cited ~~in "a"~~ above, graded approaches to Q (hence SSC ST) depend on a fully-developed risk hierarchy. Few legacy plants have those even today, in my opinion (IMO). In the 1970s, they were virtually nonexistent; everything was done as needed as you needed, often under external decision process duress.
f. The Historical Record of New Plants from the past: What is the OE that the original AEC demonstration plants saw? Around 12 were built, some lasted many years, but some were also immediately shutdown. Sheldon, NE, Atomics International's organically-moderated plant, was one of the latter, almost immediately shut down.
g. Fairness in assisting "rise-to-power" evolution (e.g., removing the Navy-style LWR reactor blinders of the past). FSV was asked to install many features[28] well after the fact to prevent Triso "fuel failure" that was (1) not define explicitly[29] and (2) LWR definition based, because nebulous regulatory requirements demanded it. IMO that should have triggered immediate shutdown decommissioning of the plant in 1985.
h. Provisions for regulatory support and alignment from Federal agencies like the Department of Energy (DOE) with vested interests in the success of new technology. (DOE/AEC was a big player in HTGR technology; their support didn't go far enough, in my opinion.)
i. Regardless, NRCs focus is Public Health and Safety (PH&S). That won't change. There are still unresolved issues around how LWR bias will apply in all future reactors. Precedent has been that bias often flies in the face of science facts. "Burning Graphite" (post Chernobyl) is one example. On that basis, graphite moderated reactor licensees encased graphite materials like reflector blocks in steel drums for Appendix R fire protection, although it remains a well-known fact graphite won't burn at Normal Atmospheric pressure-temperature conditions.[30] These groundless unscientific factoids won't go away, in the face

[27] While all licensees have corrective action programs, corrective action is demonstrated by weight of the corrective action verbiage, not actual measurable performance such as reductions in repeat rework, component failures, or procedural errors. Human factors considerations in organizations like Engineering in many plants, by my estimation, were then and are still nonexistent. Big Q Quality Programs treat issues as addressed once a "CR" condition report, has been written.
[28] The main steam line break (MSLB) shutdown trip logic was fundamental to this. This ignored the licensed plant's DBA-1 total, permanent LOFC Chapter 15 Safety Analysis
[29] Fuel failure in particle fuels is an entirely different mechanism from that in a metal clad LWR plant.
[30] In fact, graphite crucibles have been used for many years to transfer molten metals such as steel in metallurgical refinery furnace applications.

of the reported "facts", e.g., official Soviet stories about Chernobyl no one ever bothered to debunk.[31]

j. New power reactor startup licenses: Utilities are not best-suited to startup new advanced reactor prototypes. Part 53 should consider other arrangements, such as joint licenses with DOE.
k. Infrastructure requirements: There should be some sort of cursory realistic devils-advocate review that looks at the infrastructure supporting the reactor power block and assures that it supports reasonable future plant needs. Things such as access for replacements, minimization of dose, storage and laydown space all contrive to make or break new plants. Plants should not be allowed to skimp on infrastructure that will lead to unforeseen dose, safety or deferred maintenance in the future.
l. Engineering Training: Engineering training as developed by INPO is insufficient to assure capabilities in critical roles. These include Failure Analysis, Equipment Understanding, and Reliability (Statistics) Knowledge. These are presumed known by engineers, but in fact are not. Having unknowledgeable, unqualified people making decisions is unacceptable in a nuclear environment.
m. INPO: INPO is tasked by NRC to address topics such as Engineering Maintenance. They have no knowledge depth the make those decisions. INPO should be audited for effectiveness just like they and NRC evaluate members/licensees – by a qualified group with no conflict of interest.
n. Major assumptions of the FSAR/DCD: For new technologies, especially, list all the major assumptions that underpin the design. Then do a realistic brainstorm assessment of consequences if they failed. For example, coolant purity (Helium Coolant purity at FSV; Atomics International Organic Coolant Shelden, Nebraska AEC pilot plant; US Navy Sodium Cooled reactors). Failure to sustain designer-based PC limits has been the bane of many nuclear projects. Inability to meet a fundamental design constraint should be addressed immediately, not later, as it usually was. A second example is the need for certain kinds of inaccessible equipment maintenance. Control rod drive parts (blades, absorbers, cables) is a second. Asking what if those items do need maintenance, how will it be provided or will it be a shutdown issue should be "what if'ed" and brainstormed to a satisfactory conclusion before – not during – operations.

2. **Non-LWRs faced in an LWR-biased regulatory and operating environment**[32]
   a. Obfuscations that reevaluated the original license (EFW, BCD, ECD and Bearing Water)[33]
      i. EFW, Emergency Feedwater

[31] Official stories about the radioactive plume source term were based upon [unforeseen] "graphite reflector fires". They ignored other petroleum-based combustibles in the dismembered core's vicinity which contributed to fuels on the reactor-turbine refueling deck. NRC reports never acknowledged this in a report that I saw concerning FSV restart at the time – the popular press was full of stories based uncritically upon Soviet accounts of the events.

[32] This draws heavily upon FSV experience. There will be temptation to discount this as historical anomalies. That would be disservice to that Operating Experience (OE). Therefore, I disagree with their exclusion.

[33] After 1984 focus by NRC, reviewers kept applying Part 50 Appendix A criteria (LWR criteria) to regulatory to reassess the license and force design changes. Part 50.49 EQ requirements to meet LWR fuel cladding integrity requirements were the most egregious. These required installation of a steam line break shutdown system of zero safety value at the cost of around $10 million and a year to design that had absolutely no risk benefit, in my opinion.

    ii. CD, Condensate
    iii. Boosted CD, Emergency CD and Bearing Water, including BUBW

b. Interpretations of FSAR language originally more GAT/GGA sales pitches than necessity
    i. Circulator circuits
    ii. Loop shutdown logic
    iii. Circulating activity ΔΔ [end of discussion 8/17/2022]
    iv. Primary coolant noncorrosive qualities (Pure dry Helium) rarely existing. This introduced SCC and other corrosion invalidating assumptions in the FSAR

c. Imposition of LWR requirements beyond the plants licensed basis after the fact (Fuel issues – kernels, graphite cracking, EQ to MSLB requirements at FSV not in DBA-1 total loss of forced circulation. (These arose out of ex post facto application of LWR MSLB and fuel restrictions never applicable to HTGRs, IMO. These happened in 1985-6 to support restart, and in fact, increased plant costs so much the utility in hindsight would have save around $500 million for its rate payers.)

d. Discovery of new requirements not specified in license documents that affect plant equipment operational safety (PC purity shutdown, leaks, PCRV Tendon operability)
    i. Stress corrosion cracking materials upgrade to Inconel 600
    ii. PCRV design questioning, post-Chernobyl. Ignoring the structural redundancy of load redundancy for the HTGR pressure vessel tendon design, after some failed buttonheads were found in tendons. Numerically, about 10 (of ~500) tendons; Around 3 raised buttonheads (of 144) were averaged on those tendons with damage. JCOs ad infinitum attempted to explain the remaining structural integrity of the overall system was not significantly lost in any measurable regard. Load tests backed this up.

e. Control Design and Reactor Protection System (RPS)
    i. FSV general controls interfaced fully automatic controls with their Reactor Protection System (RPS). The control logic suffered two weaknesses. First was a combined General Transmission Logic for the RPS reactor loop shutdown and trips. Second was a two out of three (2/3) trip logic. This combination generated repetitive multiple "spurious" trips from cable noise voltage spikes. Use of operator hand-held transmitters was one source of these trips.[34]

3. **Programmatic developments that evolved over forty years**
    a. Absence of direct failure specification for SSC whose failure could influence the safety performance of the design. Current practice is to vaguely specify "could" but not invoke specific requirements to demonstrate "will" affect safety performance of the plant by failing a critical safety feature required. Without such criteria, many more items of SSC, materials and activities become candidates for SR classification. I suggest the simple criteria excerpted from ASME standards (ASME-RAM-1) that requires "will directly affect (fail) safety functions in failure, where that failure is a dominant failure credibly seen statistically over time."
    b. ALARA (trumping technical issues in many instances until crisis shutdowns loomed

[34] This is a gross oversimplification of a very complex "Loop Shutdown" and Reactor Trip Logic.

c. Cost compliance presumption basis (we spent x dollars to comply, therefore we are in compliance.
d. Complacency with past practices and methods and short-shrift-ing innovation
e. Maintenance Program development, including "PM" (Scheduled Maintenance)
    i. Scheduled Maintenance with respect to Equipment with Lifetimes
    ii. Understanding and use of DFM to base PM replacement
    iii. Formal recognition of life extension in mild environments
    iv. Methods of assigning intervals other than specified for inspections and overhauls
    v. Conditional overhauls
    vi. These should be addressed in the RAP portion of the DCD with some basic requirements. PMs on SR SSC should be treated as SR ST, and that decision process limited to qualified engineers (identification of tasks, type of failures – aging, random etc., life extensions for surveillance, and safe-life limits for part replacements.
    vii. Relevance of VTM-recommended maintenance versus actual implemented maintenance (VTM requirements are based on nominal 24/7 service, inappropriate for measuring aging in standby service. Substantial amounts of safety related equipment are in standby service.[35]
    viii. There are established approaches to specifying scheduled maintenance tasks. These need application amidst a specified engineering maintenance schedule, which is based upon failure mode aging lifetime, failure characteristics (aging through random failure patterns) and design strategies and known effective tasks. While VTM must be considered in the selection of tasks, other factors such as service and environment must be equally factored in.
f. Execution of licensed maintenance beyond Technical Specifications
g. Compliance demonstration by dollars, not effectiveness
h. Non-qualified Operations management interjections in technical areas and their consequential decisions
i. Unresolved issues
    i. Methods of procurements developed and simplified over 40 years
    ii. Use of statistics to evaluate SR equipment failures and identify corrections
    iii. BOP problems as they relate to nuclear NSSS side (sloppiness in maintenance program or operations)
    iv. Realism on the regulator's part dealing with new technology "non-problems"
        1. These reflect the projection (transfer) of the identified "problem" into another context requirement, exclusive of technical risk or license considerations related to that license and or technology.
    v. Licensee self-imposed requirements on the assumption of interpreted NRC directives

[35] Some nuclear plants licensed in the 1980s-1990s were committed to literal applications of VTM-recommended intervals. Others just developed this out of ignorance or fear. In both cases it was inappropriate. Initial plant programs need to be based on engineering analysis, based on the application's risk, service (duty cycle) and environment. All factor into aging life failure and determine appropriate times for part replacements and equipment service.

vi. Vendor technical materials and how those are applied in practice
vii. Commitments, with respect to reevaluating and updating obsolete or irrelevant (in a changed context) commitments
viii. Management procedural effectiveness and how that related to doing other work
ix. INPOs qualifications for roles the utility takes credit for, like Engineering Performance or Training. (How does the NRC licensing process assure they work?)
x. Technical Specifications during shutdown periods: these should cover all possible operating periods or be demonstrated to have no reasonable influence on Operations, including Maintenance Performance and Equipment Aging. (See comments on Technical Specifications purity at FSV during shutdown/depressurized non-power operations. Absence of shutdown technical specifications supported lack of concern for the shutdown in-core operating environment. These allowed PC impurities moisture to attack critical components, especially CRDs, causing SCC over time, forcing their premature rebuilding.)
xi. Misdirected NRC regulatory efforts into low-risk areas away from the substantial technical issues that the plant actually faced. (Chernobyl graphite combustion issues, Bearing Water /Backup Bearing water, Graphite fuel element cracking and failure to allow simple risk consideration on systems with multiple backups (like EFW, Boosted Condensate and BUBW) to work around minor issues learned during plant startup that should not have been allowed to shut down the plant.
xii. Overdependence on INPO for management and technical support areas (Engineering, Training, RP) performance assessment.
xiii. Ongoing use of PRA and expert panels to evaluate the risk of all potential maintenance. Restrictions should be already evaluated and covered by tech specs. PRA calculations and expert panel risk assessments should not be required for operations.
xiv. Spent fuel storage issues

j. Software issues
   i. Supporting software like CMMS systems. Maintenance Rule (50.65) compliance, Tagout
   ii. Documentation software
   iii. Drawings
   iv. Conversion of legacy blueprint software to digital forms
   v. MEL maintenance software
   vi. Scheduled Maintenance and Surveillance Program Basis software
   vii. FSAR/DCD software

k. Utility industry compliance bias
   i. Recognize that industry tends to over comply with regulatory requirements, but which I mean "over complies" by going well beyond vague idealistic specifications.
   ii. In this scenario, "money spent" is the determinant for compliance. This leads to arguments based on spend not actual performance specifications.

l. Facility Safety Programs (including 50.59-like reviews)

i. Need to be simple, clear wording (not like those based on Nuclear Energy Institute (NEI) documents or 50.69, SSC Categorization[36], which are complex and unclearly worded.)
ii. NEI guidance on using the existing 50.59 rule is obtuse. Most users, e.g., engineers, don't clearly understand it in operating nuclear plants. Compliance reviews by checklists are ineffective for the intended purpose and serve more of a rote demonstration to document than coherent analysis work. Some randomly performed 50.59 evaluations based on checklists have virtually no value than to demonstrate a good faith attempt at compliance.
iii. Facility SPs need to be numerically, statistically based on real risk
iv. Program improvements need to be broken out separately, somehow. Improvements need some kind of specific special consideration. Existing programs don't make measurable improvements, they add complexity and obtuseness in the overwhelming number of cases; that is undesirable and counter to other QP programs experience. (Simpler is more understandable, executable and preferred.)
v. Performance criteria should demonstrably go back to real events and occurrences, including designs, and show that they improve effectiveness. Process performance should be included. Since rote checklist reviews permeate the industry, whereas in my opinion, all mostly ineffective to actually define needs contribute to performance improvements, alternatives to rote checklists should be preferred.
vi. Similar comment applies to new reactor designs training. IMO, multiple guess exams are terrible. The Hon Admiral Hyman G Rickover agreed. There is a need to get training back to short answer, essay and problems solving and away from multiple guess format.
vii. Complexity: An intrinsic goal should be to identify and remove non-essential complexity. This contrasts with historical precedence and will be difficult to allow, yet it must be a condition. Failing to do so will unnecessarily burden the plant with excess costs for its lifetime.
viii. Failure progression of SSC: Require identification of a plausible failure progression where an SSC is presumed (like in PRA) to fail. Failure must be credible at the SSC level, and that must be traceable directly to a safety function or related "top" level failure. Expert panel engineers should agree. PRA can later confirm or validate risk, but the basic inputs must be fact-based, not heuristics.

m. Part 52 as a solution to Part 50 Shortcomings; Part 53 Advanced Reactor Thoughts
i. Framework A & B are both acceptable to me. I like many features about each.
ii. As suggested by Rani Francovich and Prasad Kadambi, I find non-specific approach regulatory approaches attractive, if difficult to apply practically. One way to do this would be to provide the licensee suggested areas they must address, aligned similar to the 10CFR53 Objectives Hierarchy, in the ANS Community of Practice (COP), August 26, 2022, by Rani Francovich. Then let the licensee develop their design's requirements that assure compliance with their design's risk profile. Even if this is done, some historically overlooked areas will remain that way. Let me suggest

[36] NEI Documents like NEI 18-04, purport to be the same as consensus standards prepared by ANSI-certified non-profit SDO organizations such as ASME, IEEE, and ANS.

Maintenance Programs application and, Engineering Training practical training as two.

4. **Cases:**
   a. FSV
      i. **Control rod neutron absorber** (~15-year absorber life, 40-year plant life.)
      ii. Storage capacity for unforeseen irradiated equipment and wastes (limited)
      iii. Waste and Nuclear Fuel issues beyond licensee control (DOE fuel contract, Idaho, Spent Fuel)
      iv. Irrelevance of FSV FSAR and license after 1984 control rod failure event/ordered shutdown
         1. Total Loss of Forced Circulation DBA-1 (failed a limited fraction of fuel)
         2. DBA-2, offset rupture of PC purification letdown line with respect to freeze-up
      v. **Chernobyl event** application and associated license startup delays (unscientific basis contention "burning" of graphite moderator presumed (NRC) as reported by Soviet authorities. Such graphite burning was never credible, and long since discounted.)
      vi. **Ignoring fundamental graphite properties** when alluding to LWR requirements applied post license, 1985-6. (The utility got into a quandary trying to restart the plant. Eventually over $300 million (1985 dollars) were spent to limited avail.
      vii. **ALARA trumping planned CRD PM** leading to crisis CRD rebuild 1985-6. (over 98% of all plant exposure over its life resulted from this.)
      viii. **Fluctuations:** The phenomena of fluctuations (region-based graphite column shifts under varying core differential pressures) discovered during rise to power was resolved with the addition of Region Constraint Devices or RCDs. The overall core shift remained a constraint until the end of operations, during which Appendix R concerns took over and limit power on other considerations. These reflect the risk that new reactors will have for unforeseen phenomena, which will nonetheless pose barriers to full-scale operation in a rise to power license. This makes the case for DOE participation in developing all new design reactors, or something like it.
      ix. **Emergency Planning:** Limited release fractions from severe accidents that were unrealistic increase nominal regulatory burden like STAs, etc.
      x. **Lack of performance-based criteria use** during operations to dispel many fuel failure concerns. Fuel failure definition for Triso-type coated particle fuel was never fully accepted with regulatory licenses
      xi. **D circulator Shutdown seal failure/SCC**: Evidence of He leakage issues in "D" Dog circulator (e.g., PC Pump Compressor) unresolved 6 years (shutdown seal SCC)
      xii. **CRD SCC** in control cable due to primary coolant moisture during shutdown (when specifications for moisture did not apply)
      xiii. **PC impurity technical specifications** were incomplete in that they did not consider shutdown periods.
      xiv. **Cleanup capacity** (for rise to power after shutdown) for gas reactors: FSV had grossly inadequate cleanup capacity to maintain PC helium purity during all operational periods, but especially shutdown/depressurized

xv. **Reactor Safety Logic** involving a "loop shutdown" partial loop isolation and reactor trip, intended to allow continued power operation with one loop in the face of a loop shutdown trip. In practice, this was never allowed; it was too complex to manage, and in any event a full shutdown would be needed to establish required operating chemistry and other conditions. The plant's control logic needed to be rethought and redesigned with that in mind. Because of licensing issues, it never was.

xvi. **Unforeseen events** forced more maintenance due to aging from corrosion than designers ever began to anticipate. Had there been more maintenance plan development, this would have forced those plans to change. In some instances, there should have been more realistic acknowledgement that conditions uncorrected, like PC moisture, forced the FSAR conditions to differ. More timely acknowledgement would have forced their correction or earlier termination of operations. The moisture conditions were correctable, in my opinion, with everyone's participation.[37]

xvii. **Unresolved unanticipated issues** never directly resolved at the time identified. PC Chemistry; Secondary Chemistry effects on PC chemistry from tube leaks, PC Circulator bearing water system upsets and PCRV sidewall tube leaks.[38]

b. BWR

i. **Blade tip cracking** (BWR ~20-year blade life; 40-year license life)

ii. Non-NSSS side plant trip (MPT bus duct fan)

iii. **Primary coolant system penetration IGSCC** welds repair (Mark 4 BWRs).

c. PWR

i. **PM** Program (large numbers of standby equipment in Safety Systems)

ii. **Maintenance Rule Program** – complexity issues

iii. **Obsolete** outlived early in life maintenance **commitments** persisting over time. Most of these were highly prescriptive and eventually obsolete. Inability to review and update commitments internally meant that utilities neither complied nor requested relief.

iv. **EQ Program** (rescheduling EQ PMs; demanding recalculation; commitment on penetration seals in JB in HELB areas, then not doing it. Failure to renegotiate the commitment then dumping it on me to do with virtually no support.

v. Management PMs technical decision process (lifetimes supported; those making the decisions trumped technical materials provided by reliability engineers, with virtually no credentials to do so. This illustrates why

vi. Method of evaluating outage maintenance scope; leaning into engineering to remove licensed scope by fiat (e.g., gun-decking, radioing, or pencil whipping work)

vii. ALARA

viii. Failures of MSIV for MSLB (Namco Limit Switches in High Temperature Main Steam Tunnels and related moisture-driven failures leading to MSIV trips)

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[37] I based this on similar program improvements with focused efforts over time. Moisture events in the PC were often cited as an intractable problem; they were not. They had to be closely monitored, though.

[38] These issues will accompany any new technology startup plant, IMO

ix. Overcontrol of engineering technical areas; supervision and management technical competence presumptions, when they clearly had no specific knowledge in the subject matter areas at all.
x. Absence of differentiation in staff qualification and competence. Presumptions that untrained managers would have the technical qualifications to supersede trained, qualified subordinate decisions.

5. **Caution**
   a. Complete Self-Regulated Licensee Performance-based Approach

While I agree with the principle of Risk-informed, Performance-based, non-prescriptive approaches, in actual practice they have certain limitations. Licensees that are driven by market-performance or other financial constraints, or even individuals with incentive packages on utility boards and staff, will be tempted to put purely altruistic professionalism aside. In my commercial experience, I've seen short term needs like meeting Public Utility Commission (PUC) cost and performance goals (in the case of FSV) place significant barriers to making informed decisions. Similar things are happening elsewhere, even today. Consider Georgia Powers struggles licensing their new units 3 &4 at Vogtle. The recent events at Boeing (Licensing and then grounding the Boeing 737 Max) simply reiterates this temptation. To the degree the NRC places trust and faith in Licensees to "do the right thing", they must also be circumspect at monitoring what licensees actually do. As with the Securities and Exchange Commission in the 2008 financial crisis, 20/20 vision is perfect in hindsight. No one ever understood the risks of derivative financial products until that market collapsed in 2006-7. The NRC must maintain licensee compliance oversight of ethical and altruistic license performance, to the degree they place direct faith and trust in licensees. In memory of former President Ron Reagan, "Trust, but verify."[39]

b. Licensee Maturity

Those licensees with more nuclear technology experience and maturity will have better understanding of the inherent risks in nuclear operations. Mature licensees are better prepared to take on new, advanced reactor RI-PB non-prescriptive approaches. It is improbable that new licensees will fit that mold. For other licensees, it will be ideal if the technology can be formulated to demonstrate no unreasonable public health and safety risks. I believe that there will be a need to discriminate licensees' qualifications on that basis. Certain types of plants would be more suited to less "trustworthy" (for lack of a better term) licensee operators than others. NRC should plan how it will differentiate between licensees, and which events will trigger licensee capability reviews. It is easy to see in hindsight how we could have done things differently. In the heat of the moment – public, political, cost performance, and stockholder performance pressure, doing the right thing at the time isn't always easy or clear. There are few altruistic naysayers (or even whistleblowers) to some of the wild schemes I've heard over the years. Pressure to make targets is worst among the managers who are climbing the corporate ranks or answering to Wall Street. If NRC intends to put more trust solely in the licensees, it will

[39] Strategic Arms Limitations Plans SALP), 1980s

need to rethink indicators of substandard faithful professional performance, should such events occur.[40]

Terms and acronyms

ALARA – As Low as Reasonably Achievable. A Radiation Protection program of ill-defined compliance goals with respect to cost. This non-specific program is, in my opinion, impossible for any entity to demonstrably fulfill. It should be a top regulatory/DOE program, again in my opinion, to reduce to specified goals and their related dose, time and dollar cost equivalents. As it stands now, it means RP organizations at Nuclear Plants must approve all decisions involving dose. This program drives nuclear costs upward, without limits.

ANS – American Nuclear Society

ANSI – American National Standards Institute. A standards development organization (SDO) certifying agency

AR – Advanced Reactor

ASME— American Society of Mechanical Engineering

BUBW – (FSV)Backup Bearing Water, where Bearing Water (BW) provided the lubrication water to operate the Helium Circulator compressors, analogous to a PWR RCPs or a BWRs recirculation water jet pumps

BWR – Boiling Water Reactor

CD – Condensate (including Boosted CD, Emergency CD)

CR – Condition report -- an ad hoc, indiscriminate QA report identifying any deficient condition reportable to the Quality Program, overall. At some plants these are used for other than Quality Program deficiency identification purposes as well.

DBA – Design Basis Accident

DOE – Department of Energy

EFW – Emergency Feedwater

FHM – (FSV) [nuclear] fuel handling machine; functionally similar to a LWR FHM; design-wise, not.

FSV – Fort Saint Vrain (Nuclear Generating Station)

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[40] Think of Boeing's pressure to suppress candid remarks from all those who worked and then certified the 737 Max, with its flawed, safety deficient MCAS system. Two lost airframes and nearly 400 lives later, management was still fighting a self-imposed grounding suggestion. The FAA leveled over $2 billion in fines against Boeing for misleading certification statements, but to my knowledge not a single Boeing officer on their board was ever tried for pushing their license through in 2015-2017, in the face of this knowledge. Mark Forkner, Boeing's 737 MAX Chief Technical Pilot, was the only person to face criminal charges for flaws that resulted in two fatal crashes of one of Boeing's most important planes. [NYT Mar 23, 2022] Forkner was charged with two counts of fraud involving aircraft parts in interstate commerce and four counts of wire fraud. On February 8, 2022, the Court dismissed the aircraft parts fraud charges. [www.justice.gov – Fraud Section May 2, 2022 Internet search]

GA (GGA, GAT…) – various names for the same entity over time [Gulf] General Atomics [Technology]

HTGR – High temperature gas-cooled reactor, virtually always graphite-moderated. Fuel particle design and coolants may vary, but particle fuel has been another little-understood variant to the LWR crowd.

IMO – in my opinion

INPO – Institute of Nuclear Power Operations

ISO 9001 – a process-oriented quality assurance program approach requirement

ISO International Standards Organization

LWR – Light Water Reactor

LOFC – Loss of Forced Circulation (LOFC). In the FSV HTGR, total permanent LOFC was covered as a PH&S event as DBA-1.

NEI – Nuclear Energy Institute; a nuclear industry trade association that does interpretations of regulatory requirements for nuclear utilities' plants, among other things.

NSR – Nonsafety related; SSC classified as Non-Safety related or Non SR

NSRWST – SSC Classed NSR with Special Treatments on the basis or risk-informed or PRA for Defense in Depth Addendum or equivalent for indirect capacity to reduce severity of accidents

OE – Operating Experience

PB – Performance-based

PC – Primary Coolant

PCRV – Prestressed Concrete Reactor Vessel (Fort St Vrain pressure vessel)

PH&S – Public Health and Safety

PM – Preventive Maintenance

PMs – PM tasks (where "task" means an activity that addresses one specific failure mechanism); slang. This is a slang term that originated with the US Navy. Here it is an actionable task demonstrated to manage a single failure mechanism – failure mode and its cause(s).

PRA – Probabilistic Risk Assessment

PWR – Pressurized Water Reactor

Q – Quality. Big "Q" (Prescriptive) nuclear Quality specified in various nuclear applications but especially 10CFR50 Appendix B, Quality Requirements for Nuclear Plants, and little "q", process-oriented quality practices leaning heavily on the use of statistics and non-nuclear quality ("little q") solving tools, such a s fishbone diagrams. Big "Q" is highly deterministic; little "q" is performance-based, and statistical. Both use risk aspects to determine action to do, however little q uses risk in the context of risk statistics. Big Q is not as focused statistically.

RMBK – early Soviet-designed graphite-moderated pressure-tube water reactor. Chernobyl design.

RPS – Reactor Protection System

SCC – Stress Corrosion Cracking

SDO – Standards Development Organization. A standards development organization, like CTI, the Cooling Tower Institute or AGMA, American Gear Manufacturers Association. Some are ANSI-certified, like the ASME, IEEE or ANS. Many are not, however. With certification comes more certainty and confidence in the standard product. Every organization is different in the materials they provide.

SM (scheduled maintenance)

SM Task – one activity associated factually to address one type of equipment or material failure.

SM Program (equipment and overall)

SSC – systems structures and components

SR SSC – SSC classed SR based upon Traditional Chapter 15 Safety Analysis or equivalent for direct failure capacity to fail Safety Functions

SR – Safety related

ST – Special treatment, surveillance test. In this document, ST use as Special Treatment is the default, preferred way to interpret the acronym. Special Treatments are activities specified for nuclear SR and other safety program equipment that requires some kind of certification to assure its suitability for use in a safety related or safety significant application.

Ideas to review for incorporation:

Allow clear transition to deterministic Operating Requirements avoiding esoteric, inexact or ill-defined requirements not reducible to explicit Performance-based operations requirements. For example: 1) requiring consensus among inexactly defined "expert panels" to make an engineering determination needed for SSC or their parts procurements, and surveillance test (STs) requirements elimination; 2) clear requirements for engineered special treatments including

Licensee Rule Fulfillment

Recognize that where the licensee develops the explicit fulfillment performance requirements to address regulatory requirements, they will over-specify what the requirements must be. So, for example, "expert panels of diverse background to approve all materials", will become teams of over-five people with multiple degrees in engineering, including self-declared experts, and supervision-management presumed to be qualified but without any qualifications, to meet those requirements. Such inexact qualifications, self-qualification and "others" are not conducive to effective performance. Every explicit non-specific requirement identified should have actionable fulfillment criteria to evaluate (1) its need, and 2) actual fulfillment requirements.

ALARA

To demonstrate requirements such as ALARA requirements, all explicit anticipated does generating activities should be developed well enough to project dose estimate. For example, if replacing the control rod blade shield is an expected maintenance evolution over the life of the plant, that activity should be evaluated for dose and its performance schedule, so as to know the project dose from its performance with the equipment proposed to use. That design should be available to evaluate. There should be no large dose maintenance activities needed to sustain plant operations that have not been fully developed. (This says no surprises with respect to ALARA Dose, Maintenance, and other costly unforeseen, maintenance, operational, quality-risk activities.)

Cost Estimating Practice

Every program should be fully-loaded[41] with people hours and staffs to allow estimating their overall annual hourly costs. That value should be demonstrated to offset its carrying cost by commensurate improvements in dose, lowered PHS risk, and improved operational outcomes projected. Where value can't be demonstrated, those programs should be transferred to NRC or eliminated entirely. It should not be a burden to virtually license the prototype plant during startup and relicense it again over throughout its lifetime. Adjustments to the safety license of a plant should not be so onerous as seen at FSV.

Costs of FSV Operations to GA, PSCo and DOE: GA (aka GA Technologies (GAT), Gulf General Atomics (GGA): between $1-2 billion 1985 dollars. PSCo: $1 billion 1970-1988, mostly recovered from ratepayers. The unrecovered shareholder portion I estimate at around $300 million, for operations 1984-1988. DOE's costs were on the order of $1 billion, mostly for the 8 segments of Triso graphite block fuel they procured and the cost of construction of the Independent Spent Fuel Storage Installation

[41] Aka "man" or workhour loaded, in cost accounting terms.

(ISFSI), onsite at Platteville, CO in response to Idaho Governor Cecil Andrus rejection of the spent fuel, in violation of federal laws in 1991.

Performance Based Recommendations, continued

Consolidating and streamlining engineering and administrative support processes and systems is essential to cost-effective operations. Consolidate Rules and provide one acceptable way to meet them, cost-effectively, if possible. No rule should have a specific action and a demonstrated direct safety benefit. Utilities should be encouraged to review their mandatory regulatory practices across common administrative programs, consistently for their efficient, effective application.[42]

Obvious plant equipment issues during plant operations should have been solved during design, prescribing and correct maintenance to follow, beforehand. This prescriptive requirement should be left to the designer to address but assuring it has been answered should be part of the license. A simple example is equipment aging life. Any equipment with a predicted aging life less than the plant licensed life must include a replacement plan as part of the license. Operational headaches, workarounds, outage cries and other negative impacts are thereby substantially reduced.

Nuclear design and operations immaturity led to predictable issues in retrospect in both LWR and advanced reactor situations. Deferring answering important questions with plans by handwaving is not acceptable. Procedures and practices deferment for development in the future life of the licensed plant is one. Actions that will have to be taken over the course of licensed operations must be clearly, completely spelled out. Refueling methods and maintenance plans for aging equipment are one example.[43] Disposal of expected highly-irradiated known low-level wastes is another.[44] Reasonable alternative waste mitigation plans is a third.

[42] Utilities tend to create difficult, inefficient programs with tedious compliance requirements and then blame those on NRC regulatory burden. Organizations such as INPO should challenge the need and attribution for those programs. Two I know are 50.69 programs to reclassify SSC and their requirements, and 50.59 Screening programs lifted directly from NEI regulatory requirements interpretations to perform 50.59 reviews, literally transferred to utility programs to execute. Needless to say, these legalese—written Engineering documents to implement at plants are virtually impossible for an average engineer to understand, as written. I make this statement as one doing these reviews now for well over 40 years.

[43] Superficially developed, unproven equipment was somewhat common at FSV,

[44] Control rod drive blades and absorbers, for example. These cluttered spent fuel pools at some plants in the 1980s.

August 31, 2022

Notification of Public Rulemaking, (NOPR), Part 53
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Re: Docket ID NRC-2019-0062 –
Public Comments, from JK August

To whom it may concern:

I offer the following suggestions for Part 53. Background details are attached. I don't make a point-by-point review of the proposed rule, as it stands today. I support both Framework A & B approaches, but suggest considering a third alternative, which starts from scratch. I also understand no one wants five different licenses on one plant design*. (Note: Please see the acronym list at the end of Appendix 1 pgs. 16-18 if you encounter an undefined acronym.)*

My suggestions summarized:

1. Give the licensee maximum latitude within a set of top-level constraints. Within any framework, give the designer the options to make alternative proposals.[1] Within one design, use one standard licensing approach.[2]
2. Following traditional conceptual design, alternative risk-informed processes can evaluate safety design risks effectively. Methods besides Probabilistic Risk Assessment (PRA) proposed to evaluate safety risk are encouraging.[3] Others use equally effective alternatives to PRA, or that, exclusive use of PRA is exactly counter to the charge developing Part 53 (Appendix B)
3. Performance Basing (PB): Usually traditional license-required programs under Parts 50 and 52 can be readily performance-based (PB). This includes those like Licensing, Quality Assurance and Maintenance, as examples. Licensing has many legacy commitments that can readily be reviewed for obsolescence based on risk. Utilities don't though, because requesting commitment relief is laborious, time-consuming and frankly, one of the dullest things anyone can do. The last plant I was at had several hundred maintenance commitments stemming from 1980s event. Upwards of a hundred were obsolete, and no longer relevant. They had at least one recent low-level finding for not doing exactly one simple commitment. Risk-wise, the commitment tasks made would have cost millions to perform, while introducing substantial new risk. Nonetheless, their Licensing Group was unwilling to support relief request, so they delegated the task to Engineering to somehow comply – with virtually no resources to do it. This is all too typical of problems associated with licensing commitments. Resolving them would have huge safety benefits. A second example is QA….. Consider the use of Condition Reports to demonstrate regulatory compliance to report every little

[1] Fort St. Vrain HTGR (FSV) was licensed under Part 50. Post-TMI licensee ineptitude operating the plant introduced numerous political complications. Not all can be attributed to the licensee, PSCo, alone. Operating a nuclear plant presents any new licensee a learning process need; PSCo had an industry of peers learning, too.
[2] Use one licensing method per design.
[3] Writing ANS 53.1 in 2007-2012, my conclusion was that PRA was the exclusive risk evaluation approach.

condition possibly adverse to quality. A typical pant gets maybe 50 per day per unit. That's 100 per day for a two-unit plant, or upwards of 50,000 a year! These numbers are indeed staggering. However, they lend themselves to rapid numerically rendering down. Statistically, they tell a story. Similar complaints and problems crop up, over and over. Unresolved issues that persist over time tell a story. Rarely have I seen that indicator used effectively at any nuclear plant to zero in on unresolved problems. Intense use of this indicator statistically tells another PB story, but again, utilities are not in the habit of statistically evaluating these. A third example is maintenance. Virtually every nuclear plant I've worked at has substantial amounts of uncompleted scheduled PM work orders. Many have been deferred many times. Most have processes for deferrals, and they defer the same materials over and over again, every outage review. Many PM intervals are elastic, that is, they can stretch[4]; about 10% or so statistically, can not. EQ PMs are one example. In fact, utilities only complete a small fraction of the PM work they have on their CMMS programs. It varies, but typically is well under 50%. Some run as low as 15%.[5] A performance-based measure I've proposed is to measure the compliance percentage of PMs on the CMMS scheduled "books". If compliance percentage is low, it means one of two things – the PMs aren't exact ("actionable") and aren't needed, or they're putting important equipment – like EQ qualified lifetime equipment – at risk. This is another example of PB the utility could do on their own that would greatly simplify their work and lower costs, but they won't.[6]

4. Maintenance & Reliability Assurance Programs (RAP): While the current framework provides a D-RAP for design (RAP) and assigns current license practices to cover O-RAP, e.g., Maintenance Rule 50.65, EQ Rule 50.49, other guidance (50.34 Preliminary Design, 50.46, Generic Letter 83-28, Vendor Technical Information use and maintenance….etc.) there has never been specific guidance or request for plants to pre-develop their maintenance plan based on Engineering best practices, methods or standards.[7] Such information is well-developed at this time; the methods are available to do that work better than ever in the past. Without prescribing how, any rule should require that a licensed plant predevelop their complete estimated[8] maintenance plan before the onset of operations. No plant in the US, indeed probably the world, has ever had such a requirement. The primary benefit of having such a plan is to "hit the ground running" by not having to develop maintenance activities and equipment on the fly, in crisis, after operations begin. Such plans would include addressing plans for servicing highly-irradiated equipment before it becomes "unserviceable" by being highly irradiated, with few or ineffective plans on how to manage it. This was the case for over 60 years beginning not only at Fort St Vrain (FSV), but BWRs and PWRs that later had to grapple with how to manage such tasks. Not having such plans with their bases supports continued ad hoc catch as catch can maintenance that drives high cost.

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[4] Rarely are general inspections or instrument calibrations for drift taken out to appropriate intervals.

[5] The last plant I was at had around 18,000 PMs for 2 units, 12,000 active. Every outage around 1000 PMs were tasked to systems engineering to decide whether to do or not. Literally hundreds of hours of engineering time went into the same reviews every outage, for which few if any had any specific reliability training to assess – most of whom were direct out of college. This represented a huge opportunity for the utility, yet one they could not address and take advantage of. Engineering turnover, no surprise, ran well over 20% per year.

[6] The implications here are obvious. Substantial amounts of utility scheduled maintenance have little if any benefit as scheduled. A PB approach would be reschedule, replan, or somehow figure out how to get absolutely indisputably necessary work done – like EQ PMs.

[7] This will be contested; some will argue later plants had technical specifications requiring it. They did not.

[8] Guesstimate, most probable, expected level of replacements, inspections, tests and the methods to do them.

5. Performance-based bias: Accept scientific and engineering facts and their bases. Avoid endorsing politically-based, biased third-party reports and interpretations unsupportable by factual data. I allude to issues that set the world against high-temperature graphite-moderated reactors HTGR in the 1980's and 90's post Chernobyl. The controversy over burning graphite, unscientifically based, courtesy of Soviet bloc investigators was unsupportable by physical chemistry published data. Nevertheless, NRC put a hold on FSV startup.[9] Neither reflects well on use of Engineering or Scientific practices. Jointly, these issues and many others refocused substantial efforts away from engineering where they were needed correcting real issues onto irrelevant, inconsequential low-risk distractions. Those impacts were substantial contributions to ending the pilot plant's life. I would like to present myself as a PB expert, but I'm not. I am a keen observer of processes though, with enough experience to think them through. Many utility staff are not, nor do they seek to risk challenging practices as they exist. PB-ing regulations could stimulate this needed improvement.
6. Risk-informing work: the single biggest opportunity I see to risk informing work is to establish a clear guide to what a critical risk is. This involves two aspects of work, the plants risk-based/informed design, including redundancies, etc. and the plants partition down to critical parts. This in turn requires an effective use of the term critical. Critical must be exactly defined; vaguely having the potential to affect a safety system has no potential to make an item critical. To be critical, something must have the certainty, in its failure, to directly fail the critical item one level up. While this sounds like the single failure criterion, it's not. To not explain further, see ASME-RAM-2 (2016). With a defined criticality and a plant partition hierarchy based upon safety role from the critical system design level down to the SSC part and its failure modes and causes, risk informing all work is surprisingly easy. The key requirement is to specify critical and the design basis of the plant, including all redundant safety systems that are expected to be in service. "Direct failure" then becomes the key criteria for ranking all work activity. Risk ranking to support this is rarely found in plant documentation of Generation ½ nuclear plants – those built in the 1970s-1990s. Engineering must develop it. I have spent a career risk ranking nuclear equipment in the past. This broad activity extends well into process support activities. Properly understanding risk allows the parking of substantial amounts of work that needs only to be performed when required, based on evidence of other equipment or systems failures. Doing this well takes the randomness and uncertainty out of performing tasks. SDOs would do well to provide a standard for risking ranking, on this basis.
7. Utility performance: Placing more accountability in the hands of design developers and users increases risk to the NRC. Utilities can be fundamentally not qualified to make some decisions by their governance practice or simple professionalism.[10] An example is authorizing unqualified managers and supervisors to make technical decisions in the face of those already made by qualified designers and engineers. This has been a practice in crises throughout my commercial career. It flies in the face of PE certifications, technical degrees and simple common sense – technical expertise which engineers provide – in principle. Professionally, engineers must disqualify themselves from performing work on tasks they are not qualified to do. When utilities do stupid things, which they do on occasion, they need to be taken to task. In my career I've never seen any organization except the NRC do so. By stupid things, I mean things such as issuing expired materials

[9] In the light of Chernobyl RMBK dismemberment, another would be alleged FSV PCRV construction weaknesses. Ignoring PCRV pressure vessel load redundancy, the facts did not support these issues in any way. Nonetheless they contributed to shutdown orders, raised costs, and encouraged project abandonment.
[10] This issue is not specific with regards to PSCo/FSV; it pertained most to my last direct employee engagement.

from storerooms, or rescheduling "hard time"[11] EQ work beyond scheduled work dates, demanding extensions or that engineers fabricate calculations unsupportable by scientific plant basis, calculation or facts. Usually, these activities are minor in risk, but occasionally they're not. Over time they add up. Organizations that can't live by engineering processes, shaving margins, do not deserve their license; they smack of the recent 2018-2021 Boeing 737 Max certification fiasco. This was fraught with MCAS[12] design inconsistencies and failures. (Note that after the fact, suppression of recommendations passed up – by managers – came to the public's attention for review.) Such practices must be censured at the management level, if indeed they come to pass ever. NRC alone can awaken management, and must be prepared to do so, as it has in the past.[13] It must continue to do so whatever licensing approaches are used. Simple rules and regulations make them easier to understand and enforce.[14] Simpler, more accountable rulemaking should become the complementary factual side of process, management shortcomings.

8. ALARA: As Low as Reasonably Achievable needs to be put onto more common playing field. It has driven up cost and substantially raised risks by enticing Owner operators to put off maintenance on irradiated equipment that could have been done more simply before that equipment failed. AT FSV, this included Control Rod Drive Assemblies, Circulators (RCP Pump equivalents - compressors), refueling equipment, and other equipment. Having HP (or RP, the modern name) decide who will do what maintenance when, holding the trump cards, is not good operating practice. As a maintenance professional, I find begging RP (non-engineering, non-maintenance) staff to do control rod drive PMs preventive maintenance specified totally unacceptable. This has been the case at every nuclear plant I've worked at. I can provide the list, and its close to 10. If maintenance will not be allowed above certain doses, that needs to be in the license. Period.
9. Institute of Nuclear Power Operations (INPO): NRC delegates much responsibility to INPO. Thus, they should be subject to formal review. INPO has sponsored ineffective unsound engineering practices unchallenged;[15] they are not an engineering organization. How INPO fits into the license picture needs separate review consideration. Grandfathering all INPO rules and requirements should not be assumed for new licensees. Ideally, safe plant designs would limit requirements like INPO.

Respectfully,

JK August, PE
706-834-2997
Jkaugust100000@gmail.com

Attachments

Appendix 1: Technical Supporting Points

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[11] Unextendible; e.g., must be done before the end of its life. Lifetimes are a vague notion to some unqualified staff. That's why nuclear plants have qualified engineers, buttressed by rules.

[12] Maneuvering Characteristics Augmentation System (MCAS) designed and certified for the 737 MAX to enhance the pitch stability. Ultimately certifying this system Boeing overruled numerous problems and recommendations by pilots & engineers identified during the MCAS development process.

[13] One reason I take most NRC excesses in stride is because the utility operators in question brought them on.

[14] As simple as it sounds, I have explained the concept of expired material lifetimes over many times to licensed nuclear operators and managers, to no avail. There is a point where people must simply follow the rules.

[15] One was/is their arbitrary PM tasks (PMs) extension allowance in their maintenance documents. Life limiting PMs can't arbitrarily be extended, as suggested in their materials.

Appendix 1: Technical Supporting Points
New Advanced Reactor (AR) Technology Reactor Licensing Issues for thought

Background:

My nuclear involvement has spanned 48 years in multiple reactor technologies. Since 1974, this has included naval PWRs, the Fort St. Vrain high-temperature gas reactor (HTGR), 1974-1995, GE Mark 4 and advanced BWRs, and Westinghouse and CE PWRs. I've performed shorter duration nuclear, fossil utility and transmission distribution reliability projects as well.[16] Much work has centered on developing solutions to equipment problems affecting operations and maintenance. Other efforts focused primarily on engineering administrative support process consolidation streamlining systems to be more effective. This has been a substantial utility need across the board, over my career.

Ironically, one lesson learned was that difficult nuclear engineering (re)design issues during plant operations could have been easily solvable, prescribing correct maintenance beforehand to follow.[17] Engineering redesign in lieu of maintenance is many times (~100) as expensive a way to address a problem.[18] Effective maintenance planning also substantially reduces worker exposure dose. Operational headaches, workarounds, outage cries and other negative impacts are substantially reduced. At any rate, real events from my experience support these comments. Please contact me if you have any questions.

Additional Context:

Concerns that I remember as being especially impacting follow below. These cover emerging advanced reactor technologies like General Atomics HTGRs, as well as mature LWR technology. In both LWR and advanced reactor situations, the immaturity of early nuclear generation design and operations in the 1960s, 1970s and 1980s led to issues that in retrospect were predictable. Had they been considered earlier in design phase, a significant number of these issues could have been more effectively addressed.[19] This applies especially to maintenance of highly irradiated in-core nuclear equipment like control rod absorbers. To develop NOPR comments supporting Part 53, some defining events are provided. I subcategorize issues as (1) ones that affect all new nuclear plants, (2) issues faced unique to new non-LWRs AR designs that ideally technology neutral licensing would further address and manage to develop consistently[20], and (3) other issues that all plants face, regardless of technology, engineering philosophy maturity and specific operations approach. Fort St. Vrain (FSV) had a Commercial License "rise-to-power" license. Each step of the license (stepwise power rise to 100% full 860 MWth rated thermal power) depended on successfully completing the previous steps. Once problems arose, step completion success was contentious or impossible. As one result, the plant never operated

[16] One to six years.

[17] That is, an effective maintenance program plans would have enabled more adequate maintenance performance on items that were later submitted for redesign design changes (DCs) to address. This lesson applies to all reactors, but especially to first of a kind reactors like FSV.

[18] I base this statement on initiation of design changes and eventual proposed DC review of costs.

[19] Many were not and could not be addressed, because the design wouldn't allow it. Designers had presumed a dry Helium environment for many parts that eventually failed due to cracks from fatigue and SCC cracking.

[20] Address fairly, based on that technology's licensed risk, not another technology's risk -- like LWRs. Fort St. Vrain (FSV) routinely ran the LWR litmus test, inappropriately, in my opinion. I cannot list enough instances here.

commercially at full original design rated power for any length of time.[21] Instead, holds at 30%, 40%, 60% and 84% limited it to a fraction of the original design's 860 MWth rated thermal power at various points from 1976 to 1988 shutdown. For a brief two-week period in late October 1981, the plant operated at full-rated thermal power of 860 MWth with valve wide open (VWO) [22] design generation of 330 MWe net. Many operators claimed it was the smoothest operating period of the plant.[23] While that may be a stretch, there were few unique full power operating issues then that I remember.[24]

1. **All new nuclear plants**
    a. Licensing Hierarchy (safety risk progression down to materials and their specific special treatment (ST) requirements). Specifically, many plants wound up with huge safety systems that only had one or two safety functions that required SSC designated SR. Lack of technical expertise and SSC designer categorization classification meant many bought all SSC within those systems to the highest certified safety treatment standards. NRC inspectors routinely challenged any use of NSR components bought commercially without certifications to only commercial grade standards for nonsafety related SSC. Many components and systems at FSV fit this mold, greatly increasing costs of operation with custom replacement parts. FHM and BUBW systems at FSV were notable in this regard.
    b. First plant prototype application – design demonstration requirements (rise-to-power license with NRC power rating holds). Ideally no commercial power plants would be licensed in this manner; they would be proven as viable commercial technologies first.
    c. Quality (big "Q") in relation to Plant Reliability and Availability[25]: Emphasis needs to shift towards performance-based quality requirements. The single biggest improvements would come in (1) complete delineation of SSC types, SR, NSR and NSRWST (or other category) based on how the design license does this.[26] Then (2) SSC special treatment requirements must be utterly clear, enough to specify exact, defensible STs for covered equipment. The SSC equipment partition must be clear, from top-level safety function to lower most SSC material physical degradation failure requirement. Vague inexact non-specific (hence non-actionable ST) should not be allowed to develop later, without specific conditions for doing so. It is understood, a graded approach to QA on in-scope SSC is taken.
    d. Process quality emphasis would introduce far more potential for performance-based improvements. Commercial U.S. nuclear plants fought non "big-Q" quality initiatives tooth and nail. Unfortunately, they face extremely high costs as one result due to heavy rework

[21] For a ten-day period in 1981, it operated at 100% rated thermal power.
[22] Valves wide open (VWO): the way plants normally operate at full steam load to the turbine. (Turbine admission valves wide open)
[23] NRC gave special permission exemption to restrictions for this full-power run. A "Loop Shutdown" RPS event for a false positive noise terminated it.
[24] Technical literature naysayer stories in the popular press say the plant never operated at full rated load, which is not the case.
[25] Quality (Big "Q" and quality (little "q") I use to refer to prescriptive deterministic quality like Part 50 Appendix B and process statistic quality as reflected by ISO 9000-type standards. I see more room to introduce performance based little q quality into nuclear design and operations processes. Part 53 should consider this.
[26] This is a complex subject, but better methods of partitioning SR SSC parts in SR and NSR subassembly parts is imperative to apply effective ST where they are appropriate

loads in all areas.[27] Process quality initiatives could identify and address those many shortfalls. Those "little q" quality program tools famous from Deming and Japanese use (statistical analysis of "failures", failure analysis such as Hosakawa "fishbone" diagrams, root cause analysis and proactive quality initiatives would benefit nuclear processes that are ingrained, ineffective based on statistical analysis and haven't fundamentally improved for years, witness high nuclear operating costs. While I don't believe in proscribing those specifically, I do believe that requiring demonstrated failure statistical improvements would logically move licensees toward better methods proven elsewhere. Including an annual review of licensee improvements performance might initiate efforts in that direction.

e. Graded approaches to Quality: as cited ~~in "a"~~ above, graded approaches to Q (hence SSC ST) depend on a fully-developed risk hierarchy. Few legacy plants have those even today, in my opinion (IMO). In the 1970s, they were virtually nonexistent; everything was done as needed as you needed, often under external decision process duress.
f. The Historical Record of New Plants from the past: What is the OE that the original AEC demonstration plants saw? Around 12 were built, some lasted many years, but some were also immediately shutdown. Sheldon, NE, Atomics International's organically-moderated plant, was one of the latter, almost immediately shut down.
g. Fairness in assisting "rise-to-power" evolution (e.g., removing the Navy-style LWR reactor blinders of the past). FSV was asked to install many features[28] well after the fact to prevent Triso "fuel failure" that was (1) not define explicitly[29] and (2) LWR definition based, because nebulous regulatory requirements demanded it. IMO that should have triggered immediate shutdown decommissioning of the plant in 1985.
h. Provisions for regulatory support and alignment from Federal agencies like the Department of Energy (DOE) with vested interests in the success of new technology. (DOE/AEC was a big player in HTGR technology; their support didn't go far enough, in my opinion.)
i. Regardless, NRCs focus is Public Health and Safety (PH&S). That won't change. There are still unresolved issues around how LWR bias will apply in all future reactors. Precedent has been that bias often flies in the face of science facts. "Burning Graphite" (post Chernobyl) is one example. On that basis, graphite moderated reactor licensees encased graphite materials like reflector blocks in steel drums for Appendix R fire protection, although it remains a well-known fact graphite won't burn at Normal Atmospheric pressure-temperature conditions.[30] These groundless unscientific factoids won't go away, in the face

[27] While all licensees have corrective action programs, corrective action is demonstrated by weight of the corrective action verbiage, not actual measurable performance such as reductions in repeat rework, component failures, or procedural errors. Human factors considerations in organizations like Engineering in many plants, by my estimation, were then and are still nonexistent. Big Q Quality Programs treat issues as addressed once a "CR" condition report, has been written.

[28] The main steam line break (MSLB) shutdown trip logic was fundamental to this. This ignored the licensed plant's DBA-1 total, permanent LOFC Chapter 15 Safety Analysis

[29] Fuel failure in particle fuels is an entirely different mechanism from that in a metal clad LWR plant.

[30] In fact, graphite crucibles have been used for many years to transfer molten metals such as steel in metallurgical refinery furnace applications.

of the reported "facts", e.g., official Soviet stories about Chernobyl no one ever bothered to debunk.[31]

j. New power reactor startup licenses: Utilities are not best-suited to startup new advanced reactor prototypes. Part 53 should consider other arrangements, such as joint licenses with DOE.
k. Infrastructure requirements: There should be some sort of cursory realistic devils-advocate review that looks at the infrastructure supporting the reactor power block and assures that it supports reasonable future plant needs. Things such as access for replacements, minimization of dose, storage and laydown space all contrive to make or break new plants. Plants should not be allowed to skimp on infrastructure that will lead to unforeseen dose, safety or deferred maintenance in the future.
l. Engineering Training: Engineering training as developed by INPO is insufficient to assure capabilities in critical roles. These include Failure Analysis, Equipment Understanding, and Reliability (Statistics) Knowledge. These are presumed known by engineers, but in fact are not. Having unknowledgeable, unqualified people making decisions is unacceptable in a nuclear environment.
m. INPO: INPO is tasked by NRC to address topics such as Engineering Maintenance. They have no knowledge depth the make those decisions. INPO should be audited for effectiveness just like they and NRC evaluate members/licensees – by a qualified group with no conflict of interest.
n. Major assumptions of the FSAR/DCD: For new technologies, especially, list all the major assumptions that underpin the design. Then do a realistic brainstorm assessment of consequences if they failed. For example, coolant purity (Helium Coolant purity at FSV; Atomics International Organic Coolant Shelden, Nebraska AEC pilot plant; US Navy Sodium Cooled reactors). Failure to sustain designer-based PC limits has been the bane of many nuclear projects. Inability to meet a fundamental design constraint should be addressed immediately, not later, as it usually was. A second example is the need for certain kinds of inaccessible equipment maintenance. Control rod drive parts (blades, absorbers, cables) is a second. Asking what if those items do need maintenance, how will it be provided or will it be a shutdown issue should be "what if'ed" and brainstormed to a satisfactory conclusion before – not during – operations.

2. **Non-LWRs faced in an LWR-biased regulatory and operating environment**[32]
    a. Obfuscations that reevaluated the original license (EFW, BCD, ECD and Bearing Water)[33]
        i. EFW, Emergency Feedwater

[31] Official stories about the radioactive plume source term were based upon [unforeseen] "graphite reflector fires". They ignored other petroleum-based combustibles in the dismembered core's vicinity which contributed to fuels on the reactor-turbine refueling deck. NRC reports never acknowledged this in a report that I saw concerning FSV restart at the time – the popular press was full of stories based uncritically upon Soviet accounts of the events.

[32] This draws heavily upon FSV experience. There will be temptation to discount this as historical anomalies. That would be disservice to that Operating Experience (OE). Therefore, I disagree with their exclusion.

[33] After 1984 focus by NRC, reviewers kept applying Part 50 Appendix A criteria (LWR criteria) to regulatory to reassess the license and force design changes. Part 50.49 EQ requirements to meet LWR fuel cladding integrity requirements were the most egregious. These required installation of a steam line break shutdown system of zero safety value at the cost of around $10 million and a year to design that had absolutely no risk benefit, in my opinion.

ii. CD, Condensate
iii. Boosted CD, Emergency CD and Bearing Water, including BUBW

b. Interpretations of FSAR language originally more GAT/GGA sales pitches than necessity
   i. Circulator circuits
   ii. Loop shutdown logic
   iii. Circulating activity ΔΔ [end of discussion 8/17/2022]
   iv. Primary coolant noncorrosive qualities (Pure dry Helium) rarely existing. This introduced SCC and other corrosion invalidating assumptions in the FSAR
c. Imposition of LWR requirements beyond the plants licensed basis after the fact (Fuel issues – kernels, graphite cracking, EQ to MSLB requirements at FSV not in DBA-1 total loss of forced circulation. (These arose out of ex post facto application of LWR MSLB and fuel restrictions never applicable to HTGRs, IMO. These happened in 1985-6 to support restart, and in fact, increased plant costs so much the utility in hindsight would have save around $500 million for its rate payers.)
d. Discovery of new requirements not specified in license documents that affect plant equipment operational safety (PC purity shutdown, leaks, PCRV Tendon operability)
   i. Stress corrosion cracking materials upgrade to Inconel 600
   ii. PCRV design questioning, post-Chernobyl. Ignoring the structural redundancy of load redundancy for the HTGR pressure vessel tendon design, after some failed buttonheads were found in tendons. Numerically, about 10 (of ~500) tendons; Around 3 raised buttonheads (of 144) were averaged on those tendons with damage. JCOs ad infinitum attempted to explain the remaining structural integrity of the overall system was not significantly lost in any measurable regard. Load tests backed this up.
e. Control Design and Reactor Protection System (RPS)
   i. FSV general controls interfaced fully automatic controls with their Reactor Protection System (RPS). The control logic suffered two weaknesses. First was a combined General Transmission Logic for the RPS reactor loop shutdown and trips. Second was a two out of three (2/3) trip logic. This combination generated repetitive multiple "spurious" trips from cable noise voltage spikes. Use of operator hand-held transmitters was one source of these trips.[34]

3. **Programmatic developments that evolved over forty years**
   a. Absence of direct failure specification for SSC whose failure could influence the safety performance of the design. Current practice is to vaguely specify "could" but not invoke specific requirements to demonstrate "will" affect safety performance of the plant by failing a critical safety feature required. Without such criteria, many more items of SSC, materials and activities become candidates for SR classification. I suggest the simple criteria excerpted from ASME standards (ASME-RAM-1) that requires "will directly affect (fail) safety functions in failure, where that failure is a dominant failure credibly seen statistically over time."
   b. ALARA (trumping technical issues in many instances until crisis shutdowns loomed

[34] This is a gross oversimplification of a very complex "Loop Shutdown" and Reactor Trip Logic.

c. Cost compliance presumption basis (we spent x dollars to comply, therefore we are in compliance.
d. Complacency with past practices and methods and short-shrift-ing innovation
e. Maintenance Program development, including “PM” (Scheduled Maintenance)
    i. Scheduled Maintenance with respect to Equipment with Lifetimes
    ii. Understanding and use of DFM to base PM replacement
    iii. Formal recognition of life extension in mild environments
    iv. Methods of assigning intervals other than specified for inspections and overhauls
    v. Conditional overhauls
    vi. These should be addressed in the RAP portion of the DCD with some basic requirements. PMs on SR SSC should be treated as SR ST, and that decision process limited to qualified engineers (identification of tasks, type of failures – aging, random etc., life extensions for surveillance, and safe-life limits for part replacements.
    vii. Relevance of VTM-recommended maintenance versus actual implemented maintenance (VTM requirements are based on nominal 24/7 service, inappropriate for measuring aging in standby service. Substantial amounts of safety related equipment are in standby service.[35]
    viii. There are established approaches to specifying scheduled maintenance tasks. These need application amidst a specified engineering maintenance schedule, which is based upon failure mode aging lifetime, failure characteristics (aging through random failure patterns) and design strategies and known effective tasks. While VTM must be considered in the selection of tasks, other factors such as service and environment must be equally factored in.
f. Execution of licensed maintenance beyond Technical Specifications
g. Compliance demonstration by dollars, not effectiveness
h. Non-qualified Operations management interjections in technical areas and their consequential decisions
i. Unresolved issues
    i. Methods of procurements developed and simplified over 40 years
    ii. Use of statistics to evaluate SR equipment failures and identify corrections
    iii. BOP problems as they relate to nuclear NSSS side (sloppiness in maintenance program or operations)
    iv. Realism on the regulator’s part dealing with new technology “non-problems”
        1. These reflect the projection (transfer) of the identified “problem” into another context requirement, exclusive of technical risk or license considerations related to that license and or technology.
    v. Licensee self-imposed requirements on the assumption of interpreted NRC directives

[35] Some nuclear plants licensed in the 1980s-1990s were committed to literal applications of VTM-recommended intervals. Others just developed this out of ignorance or fear. In both cases it was inappropriate. Initial plant programs need to be based on engineering analysis, based on the application’s risk, service (duty cycle) and environment. All factor into aging life failure and determine appropriate times for part replacements and equipment service.

vi. Vendor technical materials and how those are applied in practice
vii. Commitments, with respect to reevaluating and updating obsolete or irrelevant (in a changed context) commitments
viii. Management procedural effectiveness and how that related to doing other work
ix. INPOs qualifications for roles the utility takes credit for, like Engineering Performance or Training. (How does the NRC licensing process assure they work?)
x. Technical Specifications during shutdown periods: these should cover all possible operating periods or be demonstrated to have no reasonable influence on Operations, including Maintenance Performance and Equipment Aging. (See comments on Technical Specifications purity at FSV during shutdown/depressurized non-power operations. Absence of shutdown technical specifications supported lack of concern for the shutdown in-core operating environment. These allowed PC impurities moisture to attack critical components, especially CRDs, causing SCC over time, forcing their premature rebuilding.)
xi. Misdirected NRC regulatory efforts into low-risk areas away from the substantial technical issues that the plant actually faced. (Chernobyl graphite combustion issues, Bearing Water /Backup Bearing water, Graphite fuel element cracking and failure to allow simple risk consideration on systems with multiple backups (like EFW, Boosted Condensate and BUBW) to work around minor issues learned during plant startup that should not have been allowed to shut down the plant.
xii. Overdependence on INPO for management and technical support areas (Engineering, Training, RP) performance assessment.
xiii. Ongoing use of PRA and expert panels to evaluate the risk of all potential maintenance. Restrictions should be already evaluated and covered by tech specs. PRA calculations and expert panel risk assessments should not be required for operations.
xiv. Spent fuel storage issues

j. Software issues
   i. Supporting software like CMMS systems. Maintenance Rule (50.65) compliance, Tagout
   ii. Documentation software
   iii. Drawings
   iv. Conversion of legacy blueprint software to digital forms
   v. MEL maintenance software
   vi. Scheduled Maintenance and Surveillance Program Basis software
   vii. FSAR/DCD software

k. Utility industry compliance bias
   i. Recognize that industry tends to over comply with regulatory requirements, but which I mean “over complies” by going well beyond vague idealistic specifications.
   ii. In this scenario, “money spent” is the determinant for compliance. This leads to arguments based on spend not actual performance specifications.

l. Facility Safety Programs (including 50.59-like reviews)

i. Need to be simple, clear wording (not like those based on Nuclear Energy Institute (NEI) documents or 50.69, SSC Categorization[36], which are complex and unclearly worded.)
ii. NEI guidance on using the existing 50.59 rule is obtuse. Most users, e.g., engineers, don't clearly understand it in operating nuclear plants. Compliance reviews by checklists are ineffective for the intended purpose and serve more of a rote demonstration to document than coherent analysis work. Some randomly performed 50.59 evaluations based on checklists have virtually no value than to demonstrate a good faith attempt at compliance.
iii. Facility SPs need to be numerically, statistically based on real risk
iv. Program improvements need to be broken out separately, somehow. Improvements need some kind of specific special consideration. Existing programs don't make measurable improvements, they add complexity and obtuseness in the overwhelming number of cases; that is undesirable and counter to other QP programs experience. (Simpler is more understandable, executable and preferred.)
v. Performance criteria should demonstrably go back to real events and occurrences, including designs, and show that they improve effectiveness. Process performance should be included. Since rote checklist reviews permeate the industry, whereas in my opinion, all mostly ineffective to actually define needs contribute to performance improvements, alternatives to rote checklists should be preferred.
vi. Similar comment applies to new reactor designs training. IMO, multiple guess exams are terrible. The Hon Admiral Hyman G Rickover agreed. There is a need to get training back to short answer, essay and problems solving and away from multiple guess format.
vii. Complexity: An intrinsic goal should be to identify and remove non-essential complexity. This contrasts with historical precedence and will be difficult to allow, yet it must be a condition. Failing to do so will unnecessarily burden the plant with excess costs for its lifetime.
viii. Failure progression of SSC: Require identification of a plausible failure progression where an SSC is presumed (like in PRA) to fail. Failure must be credible at the SSC level, and that must be traceable directly to a safety function or related "top" level failure. Expert panel engineers should agree. PRA can later confirm or validate risk, but the basic inputs must be fact-based, not heuristics.

m. Part 52 as a solution to Part 50 Shortcomings; Part 53 Advanced Reactor Thoughts
   i. Framework A & B are both acceptable to me. I like many features about each.
   ii. As suggested by Rani Francovich and Prasad Kadambi, I find non-specific approach regulatory approaches attractive, if difficult to apply practically. One way to do this would be to provide the licensee suggested areas they must address, aligned similar to the 10CFR53 Objectives Hierarchy, in the ANS Community of Practice (COP), August 26, 2022, by Rani Francovich. Then let the licensee develop their design's requirements that assure compliance with their design's risk profile. Even if this is done, some historically overlooked areas will remain that way. Let me suggest

[36] NEI Documents like NEI 18-04, purport to be the same as consensus standards prepared by ANSI-certified non-profit SDO organizations such as ASME, IEEE, and ANS.

Maintenance Programs application and, Engineering Training practical training as two.

4. **Cases:**
   a. FSV
      i. **Control rod neutron absorber** (~15-year absorber life, 40-year plant life.)
      ii. Storage capacity for unforeseen irradiated equipment and wastes (limited)
      iii. Waste and Nuclear Fuel issues beyond licensee control (DOE fuel contract, Idaho, Spent Fuel)
      iv. Irrelevance of FSV FSAR and license after 1984 control rod failure event/ordered shutdown
         1. Total Loss of Forced Circulation DBA-1 (failed a limited fraction of fuel)
         2. DBA-2, offset rupture of PC purification letdown line with respect to freeze-up
      v. **Chernobyl event** application and associated license startup delays (unscientific basis contention "burning" of graphite moderator presumed (NRC) as reported by Soviet authorities. Such graphite burning was never credible, and long since discounted.)
      vi. **Ignoring fundamental graphite properties** when alluding to LWR requirements applied post license, 1985-6. (The utility got into a quandary trying to restart the plant. Eventually over $300 million (1985 dollars) were spent to limited avail.
      vii. **ALARA trumping planned CRD PM** leading to crisis CRD rebuild 1985-6. (over 98% of all plant exposure over its life resulted from this.)
      viii. **Fluctuations:** The phenomena of fluctuations (region-based graphite column shifts under varying core differential pressures) discovered during rise to power was resolved with the addition of Region Constraint Devices or RCDs. The overall core shift remained a constraint until the end of operations, during which Appendix R concerns took over and limit power on other considerations. These reflect the risk that new reactors will have for unforeseen phenomena, which will nonetheless pose barriers to full-scale operation in a rise to power license. This makes the case for DOE participation in developing all new design reactors, or something like it.
      ix. **Emergency Planning:** Limited release fractions from severe accidents that were unrealistic increase nominal regulatory burden like STAs, etc.
      x. **Lack of performance-based criteria use** during operations to dispel many fuel failure concerns. Fuel failure definition for Triso-type coated particle fuel was never fully accepted with regulatory licenses
      xi. **D circulator Shutdown seal failure/SCC**: Evidence of He leakage issues in "D" Dog circulator (e.g., PC Pump Compressor) unresolved 6 years (shutdown seal SCC)
      xii. **CRD SCC** in control cable due to primary coolant moisture during shutdown (when specifications for moisture did not apply)
      xiii. **PC impurity technical specifications** were incomplete in that they did not consider shutdown periods.
      xiv. **Cleanup capacity** (for rise to power after shutdown) for gas reactors: FSV had grossly inadequate cleanup capacity to maintain PC helium purity during all operational periods, but especially shutdown/depressurized

xv. **Reactor Safety Logic** involving a "loop shutdown" partial loop isolation and reactor trip, intended to allow continued power operation with one loop in the face of a loop shutdown trip. In practice, this was never allowed; it was too complex to manage, and in any event a full shutdown would be needed to establish required operating chemistry and other conditions. The plant's control logic needed to be rethought and redesigned with that in mind. Because of licensing issues, it never was.

xvi. **Unforeseen events** forced more maintenance due to aging from corrosion than designers ever began to anticipate. Had there been more maintenance plan development, this would have forced those plans to change. In some instances, there should have been more realistic acknowledgement that conditions uncorrected, like PC moisture, forced the FSAR conditions to differ. More timely acknowledgement would have forced their correction or earlier termination of operations. The moisture conditions were correctable, in my opinion, with everyone's participation.[37]

xvii. **Unresolved unanticipated issues** never directly resolved at the time identified. PC Chemistry; Secondary Chemistry effects on PC chemistry from tube leaks, PC Circulator bearing water system upsets and PCRV sidewall tube leaks.[38]

b. BWR

i. **Blade tip cracking** (BWR ~20-year blade life; 40-year license life)

ii. Non-NSSS side plant trip (MPT bus duct fan)

iii. **Primary coolant system penetration IGSCC** welds repair (Mark 4 BWRs).

c. PWR

i. **PM** Program (large numbers of standby equipment in Safety Systems)

ii. Maintenance Rule Program – complexity issues

iii. **Obsolete** outlived early in life maintenance **commitments** persisting over time. Most of these were highly prescriptive and eventually obsolete. Inability to review and update commitments internally meant that utilities neither complied nor requested relief.

iv. **EQ Program** (rescheduling EQ PMs; demanding recalculation; commitment on penetration seals in JB in HELB areas, then not doing it. Failure to renegotiate the commitment then dumping it on me to do with virtually no support.

v. Management PMs technical decision process (lifetimes supported; those making the decisions trumped technical materials provided by reliability engineers, with virtually no credentials to do so. This illustrates why

vi. Method of evaluating outage maintenance scope; leaning into engineering to remove licensed scope by fiat (e.g., gun-decking, radioing, or pencil whipping work)

vii. ALARA

viii. Failures of MSIV for MSLB (Namco Limit Switches in High Temperature Main Steam Tunnels and related moisture-driven failures leading to MSIV trips)

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[37] I based this on similar program improvements with focused efforts over time. Moisture events in the PC were often cited as an intractable problem; they were not. They had to be closely monitored, though.

[38] These issues will accompany any new technology startup plant, IMO

ix. Overcontrol of engineering technical areas; supervision and management technical competence presumptions, when they clearly had no specific knowledge in the subject matter areas at all.
x. Absence of differentiation in staff qualification and competence. Presumptions that untrained managers would have the technical qualifications to supersede trained, qualified subordinate decisions.

5. **Caution**

a. Complete Self-Regulated Licensee Performance-based Approach

While I agree with the principle of Risk-informed, Performance-based, non-prescriptive approaches, in actual practice they have certain limitations. Licensees that are driven by market-performance or other financial constraints, or even individuals with incentive packages on utility boards and staff, will be tempted to put purely altruistic professionalism aside. In my commercial experience, I've seen short term needs like meeting Public Utility Commission (PUC) cost and performance goals (in the case of FSV) place significant barriers to making informed decisions. Similar things are happening elsewhere, even today. Consider Georgia Powers struggles licensing their new units 3 &4 at Vogtle. The recent events at Boeing (Licensing and then grounding the Boeing 737 Max) simply reiterates this temptation. To the degree the NRC places trust and faith in Licensees to "do the right thing", they must also be circumspect at monitoring what licensees actually do. As with the Securities and Exchange Commission in the 2008 financial crisis, 20/20 vision is perfect in hindsight. No one ever understood the risks of derivative financial products until that market collapsed in 2006-7. The NRC must maintain licensee compliance oversight of ethical and altruistic license performance, to the degree they place direct faith and trust in licensees. In memory of former President Ron Reagan, "Trust, but verify."[39]

b. Licensee Maturity

Those licensees with more nuclear technology experience and maturity will have better understanding of the inherent risks in nuclear operations. Mature licensees are better prepared to take on new, advanced reactor RI-PB non-prescriptive approaches. It is improbable that new licensees will fit that mold. For other licensees, it will be ideal if the technology can be formulated to demonstrate no unreasonable public health and safety risks. I believe that there will be a need to discriminate licensees' qualifications on that basis. Certain types of plants would be more suited to less "trustworthy" (for lack of a better term) licensee operators than others. NRC should plan how it will differentiate between licensees, and which events will trigger licensee capability reviews. It is easy to see in hindsight how we could have done things differently. In the heat of the moment – public, political, cost performance, and stockholder performance pressure, doing the right thing at the time isn't always easy or clear. There are few altruistic naysayers (or even whistleblowers) to some of the wild schemes I've heard over the years. Pressure to make targets is worst among the managers who are climbing the corporate ranks or answering to Wall Street. If NRC intends to put more trust solely in the licensees, it will

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[39] Strategic Arms Limitations Plans SALP), 1980s

need to rethink indicators of substandard faithful professional performance, should such events occur.[40]

Terms and acronyms

ALARA – As Low as Reasonably Achievable. A Radiation Protection program of ill-defined compliance goals with respect to cost. This non-specific program is, in my opinion, impossible for any entity to demonstrably fulfill. It should be a top regulatory/DOE program, again in my opinion, to reduce to specified goals and their related dose, time and dollar cost equivalents. As it stands now, it means RP organizations at Nuclear Plants must approve all decisions involving dose. This program drives nuclear costs upward, without limits.

ANS – American Nuclear Society

ANSI – American National Standards Institute. A standards development organization (SDO) certifying agency

AR – Advanced Reactor

ASME— American Society of Mechanical Engineering

BUBW – (FSV)Backup Bearing Water, where Bearing Water (BW) provided the lubrication water to operate the Helium Circulator compressors, analogous to a PWR RCPs or a BWRs recirculation water jet pumps

BWR – Boiling Water Reactor

CD – Condensate (including Boosted CD, Emergency CD)

CR – Condition report -- an ad hoc, indiscriminate QA report identifying any deficient condition reportable to the Quality Program, overall. At some plants these are used for other than Quality Program deficiency identification purposes as well.

DBA – Design Basis Accident

DOE – Department of Energy

EFW – Emergency Feedwater

FHM – (FSV) [nuclear] fuel handling machine; functionally similar to a LWR FHM; design-wise, not.

FSV – Fort Saint Vrain (Nuclear Generating Station)

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[40] Think of Boeing's pressure to suppress candid remarks from all those who worked and then certified the 737 Max, with its flawed, safety deficient MCAS system. Two lost airframes and nearly 400 lives later, management was still fighting a self-imposed grounding suggestion. The FAA leveled over $2 billion in fines against Boeing for misleading certification statements, but to my knowledge not a single Boeing officer on their board was ever tried for pushing their license through in 2015-2017, in the face of this knowledge. Mark Forkner, Boeing's 737 MAX Chief Technical Pilot, was the only person to face criminal charges for flaws that resulted in two fatal crashes of one of Boeing's most important planes. [NYT Mar 23, 2022] Forkner was charged with two counts of fraud involving aircraft parts in interstate commerce and four counts of wire fraud. On February 8, 2022, the Court dismissed the aircraft parts fraud charges. [www.justice.gov – Fraud Section May 2, 2022 Internet search]

GA (GGA, GAT…) – various names for the same entity over time [Gulf] General Atomics [Technology]

HTGR – High temperature gas-cooled reactor, virtually always graphite-moderated. Fuel particle design and coolants may vary, but particle fuel has been another little-understood variant to the LWR crowd.

IMO – in my opinion

INPO – Institute of Nuclear Power Operations

ISO 9001 – a process-oriented quality assurance program approach requirement

ISO International Standards Organization

LWR – Light Water Reactor

LOFC – Loss of Forced Circulation (LOFC). In the FSV HTGR, total permanent LOFC was covered as a PH&S event as DBA-1.

NEI – Nuclear Energy Institute; a nuclear industry trade association that does interpretations of regulatory requirements for nuclear utilities' plants, among other things.

NSR – Nonsafety related; SSC classified as Non-Safety related or Non SR

NSRWST – SSC Classed NSR with Special Treatments on the basis or risk-informed or PRA for Defense in Depth Addendum or equivalent for indirect capacity to reduce severity of accidents

OE – Operating Experience

PB – Performance-based

PC – Primary Coolant

PCRV – Prestressed Concrete Reactor Vessel (Fort St Vrain pressure vessel)

PH&S – Public Health and Safety

PM – Preventive Maintenance

PMs – PM tasks (where "task" means an activity that addresses one specific failure mechanism); slang. This is a slang term that originated with the US Navy. Here it is an actionable task demonstrated to manage a single failure mechanism – failure mode and its cause(s).

PRA – Probabilistic Risk Assessment

PWR – Pressurized Water Reactor

Q – Quality. Big "Q" (Prescriptive) nuclear Quality specified in various nuclear applications but especially 10CFR50 Appendix B, Quality Requirements for Nuclear Plants, and little "q", process-oriented quality practices leaning heavily on the use of statistics and non-nuclear quality ("little q") solving tools, such a s fishbone diagrams. Big "Q" is highly deterministic; little "q" is performance-based, and statistical. Both use risk aspects to determine action to do, however little q uses risk in the context of risk statistics. Big Q is not as focused statistically.

RMBK – early Soviet-designed graphite-moderated pressure-tube water reactor. Chernobyl design.

RPS – Reactor Protection System

SCC – Stress Corrosion Cracking

SDO – Standards Development Organization. A standards development organization, like CTI, the Cooling Tower Institute or AGMA, American Gear Manufacturers Association. Some are ANSI-certified, like the ASME, IEEE or ANS. Many are not, however. With certification comes more certainty and confidence in the standard product. Every organization is different in the materials they provide.

SM (scheduled maintenance)

SM Task – one activity associated factually to address one type of equipment or material failure.

SM Program (equipment and overall)

SSC – systems structures and components

SR SSC – SSC classed SR based upon Traditional Chapter 15 Safety Analysis or equivalent for direct failure capacity to fail Safety Functions

SR – Safety related

ST – Special treatment, surveillance test. In this document, ST use as Special Treatment is the default, preferred way to interpret the acronym. Special Treatments are activities specified for nuclear SR and other safety program equipment that requires some kind of certification to assure its suitability for use in a safety related or safety significant application.

Ideas to review for incorporation:

Allow clear transition to deterministic Operating Requirements avoiding esoteric, inexact or ill-defined requirements not reducible to explicit Performance-based operations requirements. For example: 1) requiring consensus among inexactly defined "expert panels" to make an engineering determination needed for SSC or their parts procurements, and surveillance test (STs) requirements elimination; 2) clear requirements for engineered special treatments including

Licensee Rule Fulfillment

Recognize that where the licensee develops the explicit fulfillment performance requirements to address regulatory requirements, they will over-specify what the requirements must be. So, for example, "expert panels of diverse background to approve all materials", will become teams of over-five people with multiple degrees in engineering, including self-declared experts, and supervision-management presumed to be qualified but without any qualifications, to meet those requirements. Such inexact qualifications, self-qualification and "others" are not conducive to effective performance. Every explicit non-specific requirement identified should have actionable fulfillment criteria to evaluate (1) its need, and 2) actual fulfillment requirements.

ALARA

To demonstrate requirements such as ALARA requirements, all explicit anticipated does generating activities should be developed well enough to project dose estimate. For example, if replacing the control rod blade shield is an expected maintenance evolution over the life of the plant, that activity should be evaluated for dose and its performance schedule, so as to know the project dose from its performance with the equipment proposed to use. That design should be available to evaluate. There should be no large dose maintenance activities needed to sustain plant operations that have not been fully developed. (This says no surprises with respect to ALARA Dose, Maintenance, and other costly unforeseen, maintenance, operational, quality-risk activities.)

Cost Estimating Practice

Every program should be fully-loaded[41] with people hours and staffs to allow estimating their overall annual hourly costs. That value should be demonstrated to offset its carrying cost by commensurate improvements in dose, lowered PHS risk, and improved operational outcomes projected. Where value can't be demonstrated, those programs should be transferred to NRC or eliminated entirely. It should not be a burden to virtually license the prototype plant during startup and relicense it again over throughout its lifetime. Adjustments to the safety license of a plant should not be so onerous as seen at FSV.

Costs of FSV Operations to GA, PSCo and DOE: GA (aka GA Technologies (GAT), Gulf General Atomics (GGA): between $1-2 billion 1985 dollars. PSCo: $1 billion 1970-1988, mostly recovered from ratepayers. The unrecovered shareholder portion I estimate at around $300 million, for operations 1984-1988. DOE's costs were on the order of $1 billion, mostly for the 8 segments of Triso graphite block fuel they procured and the cost of construction of the Independent Spent Fuel Storage Installation

[41] Aka "man" or workhour loaded, in cost accounting terms.

(ISFSI), onsite at Platteville, CO in response to Idaho Governor Cecil Andrus rejection of the spent fuel, in violation of federal laws in 1991.

Performance Based Recommendations, continued

Consolidating and streamlining engineering and administrative support processes and systems is essential to cost-effective operations. Consolidate Rules and provide one acceptable way to meet them, cost-effectively, if possible. No rule should have a specific action and a demonstrated direct safety benefit. Utilities should be encouraged to review their mandatory regulatory practices across common administrative programs, consistently for their efficient, effective application.[42]

Obvious plant equipment issues during plant operations should have been solved during design, prescribing and correct maintenance to follow, beforehand. This prescriptive requirement should be left to the designer to address but assuring it has been answered should be part of the license. A simple example is equipment aging life. Any equipment with a predicted aging life less than the plant licensed life must include a replacement plan as part of the license. Operational headaches, workarounds, outage cries and other negative impacts are thereby substantially reduced.

Nuclear design and operations immaturity led to predictable issues in retrospect in both LWR and advanced reactor situations. Deferring answering important questions with plans by handwaving is not acceptable. Procedures and practices deferment for development in the future life of the licensed plant is one. Actions that will have to be taken over the course of licensed operations must be clearly, completely spelled out. Refueling methods and maintenance plans for aging equipment are one example.[43] Disposal of expected highly-irradiated known low-level wastes is another.[44] Reasonable alternative waste mitigation plans is a third.

[42] Utilities tend to create difficult, inefficient programs with tedious compliance requirements and then blame those on NRC regulatory burden. Organizations such as INPO should challenge the need and attribution for those programs. Two I know are 50.69 programs to reclassify SSC and their requirements, and 50.59 Screening programs lifted directly from NEI regulatory requirements interpretations to perform 50.59 reviews, literally transferred to utility programs to execute. Needless to say, these legalese—written Engineering documents to implement at plants are virtually impossible for an average engineer to understand, as written. I make this statement as one doing these reviews now for well over 40 years.

[43] Superficially developed, unproven equipment was somewhat common at FSV,

[44] Control rod drive blades and absorbers, for example. These cluttered spent fuel pools at some plants in the 1980s.