Union of Concerned Scientists
Citizens and Scientists for Environmental Solutions

COMMENTS ON RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS

In Section VI.2.0, "Questions for Public Input," of the *Federal Register Notice* dated May 16, 2003 (Vol. 68, No. 95), the NRC solicited public comment on specific elements of its proposed rule. UCS provides the following responses (in **boldface** type) to those specific questions:

1) The proposed rule requires as a minimum, a PRA that includes internal events, at power, which has been subjected to a peer review process. The PRA (for that scope) must be capable of determining both CDF and LERF (i.e., provide level 2-type results). The Commission is seeking comment as to whether the NRC should amend the requirements in Section 50.69(c) to require a level 2 internal and external initiating events, all-mode, peer-reviewed PRA that must be submitted to, and reviewed by, the NRC. Thus, instead of employing other methods to account for the contribution from modes and events not modeled in the PRA, this more comprehensive PRA would allow for quantification of the contribution from these scenarios. This approach would involve substantive changes in the implementing guidance as well.

   UCS Response: The proposed rule, if adopted in its present form, would allow reactor licensees to classify components based largely on the results from at-power PRAs for internal events alone. The proposed rule does not restrict the reclassifications under the proposed rule to only those components performing a function for internal events at power. It is totally inappropriate to use a limited-scope tool to make unlimited scope reclassifications. As former NRC Chairman Richard Meserve pointed out in his comments on the rulemaking package:

   > ***"The achievement of the purpose of this rule will not be possible without confidence in the quality and scope of the underlying PRAs."*** [1]

   UCS has documented many quality problems with PRAs in our August 2000 report titled, "Nuclear Plant Risk Studies: Failing the Grade" (now available in ADAMS under Accession No. ML010810228). Those problems remain unresolved and continue to plague the PRAs. Largely at the urging of NRC Commissioner Edward McGaffigan, UCS was able to observe the peer review conducted in summer 2001 of the North Anna PRA. Our observation focused on the peer review process rather than on the quality of the North Anna PRA. Our comments and recommendations on the peer review process for PRAs were documented in a report dated September 10, 2001, titled "Risk Incites: Observations on the PRA Peer Review Process." The recommendations remain outstanding.

   In November 2002, the NRC issued Revision 1 of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." This Reg Guide is relevant because it would shape NRC staff decision-making if the proposed rule is adopted, as indicated by paragraph (1) in Section 2, "An Acceptable Approach to Risk-Informed Decision-Making," which states, "The proposed change meets the current regulation...". Thus, a licensee opting for 10 CFR 50.69 handling would submit a licensing package to the NRC which would be processed using Reg Guide 1.174. Section 1.4, "Scope of This Regulatory Guide," states "The analyses should reflect the actual

[1] Nuclear Regulation Commission, Commission Voting Record for SECY-02-0176, March 28, 2003.

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design, construction, and operational practices of the plant." No US plant is designed or constructed for only operation at power, and none operates that way. The PRA used for this rulemaking should address how the plants are designed, constructed, and operated and not for some limited subset of their design, construction, and operation. As NRC Commissioner Edward McGaffigan noted:

> ***"I do not believe any baseline CDF [core damage frequency] number to be accurate to better than a factor of about 10. Yet we are proposing to make distinctions based on such numbers. ... The staff is not willing to require a good all-mode, internal- and external-initiating event level 2 PRA, but the staff acts as if we have one at times, and at other times applies band-aids to try to deal with the problems that arise in dealing with less comprehensive PRAs."*** [2]

As the NRC staff noted, "*This rulemaking is the first instance in which the NRC would establish, by rule, specific requirements concerning the conduct of a PRA in support of a particular regulatory action.*"[3] The NRC must get it right from the onset. That's the best way to meet the agency's oft-stated goals of (1) maintaining safety, (2) reducing unnecessary regulatory burden, (3) improving agency effectiveness and efficiency, and (4) improving public confidence. The NRC must establish minimum standards for full-scope, internal- and external event level 2 PRAs and verify that PRAs meet or exceed those standards BEFORE using their results to lessen regulatory requirements.

It would be regulatory folly to use a limited scope PRA, even if it were level 2 and peer reviewed, to make decisions on how to treat equipment <u>outside</u> the scope of the PRA. For example, an at-power level 2 PRA is mute with respect to refueling equipment, to some equipment needed during mid-loop operation at pressurized water reactors, etc. even though this equipment must function properly to prevent/mitigate design bases accidents such as a fuel handling accident and a loss of offsite power.

The proposed rulemaking would require an "expert panel" or equivalent process be used to reclassify equipment outside the scope of the at-power, internal events PRA. In theory, this approach seems like a viable alternative. But what prevents the expert panel from essentially blanket reclassifications of out-of-scope equipment on the flimsy excuse that if it were safety significant, it would appear in the PRA? The proposed rulemaking fails to establish appropriate expectations for the "expert panels." This failure will prevent plant owners from good faith efforts to meet or exceed those expectations and later prevent NRC inspectors from evaluating whether expert panels functioned adequately.

The limited scope PRA/expert panel approach proposed for 10 CFR 50.69 is similar to the approach used for the Maintenance Rule (10 CFR 50.65). Both approaches feature distinctions between components using results from limited scope PRAs of uncertain quality supplemented by expert panel judgments. But there is a very important difference between 10 CFR 50.65 and 10 CFR 50.69 – 10 CFR 50.65 *added* monitoring requirements for non-safety-related components meeting certain criteria while 10 CFR 50.69 *removes* requirements for safety-related components meeting certain criteria. For 10 CFR 50.65, the worst that could happen would be for the limited scope PRA/expert panel approach failing to select non-safety-related component(s) for heightened monitoring. Safety margins would not be compromised, they would simply fail to be enhanced. For 10 CFR 50.69, the worst that can happen will be for the limited scope PRA/expert panel approach mistakenly reclassifying safety-related component(s) as RISC-3 for lessened handling. Safety margins would be compromised.

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[2] Nuclear Regulation Commission, Commission Voting Record for SECY-02-0176, March 28, 2003.

[3] Nuclear Regulatory Commission Letter dated September 30, 2002, from William D. Travers, Executive Director for Operations, to Chairman and Commissioners, SECY-02-9176, "Proposed Rulemaking to Add New Section 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components,"" page 4.

The proposed 10 CFR 50.69 would apply to all equipment having "special treatment" provisions at nuclear power plants, and not just that subset of "special treatment" equipment covered by an at-power, level 2 PRA for internal events. Thus, either the scope of 10 CFR 50.69 must be reduced to apply to only equipment requiring "special treatment" from internal events at power –or– the level 2 PRA must cover all operating modes, including defueled, for internal and external events –or– the rule must be revised to explicitly define expectations for the expert panel process.

And finally, the NRC must determine the sanity of using results from even the best quality, all-mode PRA for internal and external events to justify reducing regulatory oversight of safety-related equipment. It's akin to an "A" student trying to justify cutting classes, foregoing homework, and refusing to study based on having received "A's" on his/her report card. It seems probable that going to class, doing the homework, and studying for tests is responsible for the good grades. It is therefore absurd to point to the good grades as rationale for ending the actions producing those results. Likewise, the NRC seems to want to use the results from PRAs using equipment reliability data WHEN THE EQUIPMENT WAS SUBJECTED TO HIGHER REGULATORY OVERSIGHT to justify cutting back on the oversight. Is the NRC stipulating that its past regulatory oversight had no value? If not, how can it reduce the regulatory oversight on equipment based on past performance results that benefited from NRC oversight?

2) The proposed rule requires the licensee to submit information about its PRA and these other methods, including information about the quality and level of detail about all of the methods to be used.

UCS Response: In November 2002, the NRC issued Revision 1 of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Section 1.2 of this Reg Guide states "PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review."

By e-mail dated May 27, 2003, UCS requested a copy of the Individual Plant Examinations (IPEs) and Updated Final Safety Analysis Reports (UFSARs) for one pressurized water reactor (Shearon Harris) and one boiling water reactor (Vermont Yankee) so as to be able to provide meaningful comments on this proposed rulemaking. On May 28, 2003, Mr. Timothy Reed on the NRC staff called me to inform me that my request had been denied. The NRC would not permit me access to the IPEs and UFSARs for the requested sites, or any sites.

Prior to the tragic events of September 11th, the IPEs and UFSARs were publicly available in the NRC's Public Document Room. After the tragic events, the NRC removed these documents, any other documents, from the public arena for security reasons. While some documents have been returned to the public arena, the NRC is still assessing where to draw the line on the risk and vulnerability information contained in documents such as the IPEs and UFSARs. The NRC's own SPAR models are presently unavailable to the public.

This 10 CFR 50.69 rulemaking effort must be suspended and resumed after the NRC finalizes where that line is drawn and makes relevant information on PRAs from the public side of that line available. Absent access to at least that information, the public cannot adequately comment on this important question. Without access to the UFSARs, we cannot determine the systems, structures, and components needed to prevent and mitigate design basis accidents. Without access to the IPEs or PRAs, we cannot determine where the results from limited scope, at-power PRAs fail to adequately address systems, structures, and components with vital safety functions. The NRC must not proceed with the first risk-informed rule at a time when it has removed much of the relevant information from the public arena. This is an ill-advised public policy step that is totally unnecessary – the nuclear power plants have operated for more then three decades under the existing regulatory scheme and could continue to operate until the NRC figures out what information can be made publicly available.

3) The Commission is also seeking comment on whether a different set of PRA requirements, from either of the alternatives described above, should be required for this application.

UCS Response: As in the case last year where it secretly jettisoned years of work by the Discrimination Task Force and replaced it with poorly constructed recommendations on safety conscious work environments, NRC senior management has undermined confidence in the regulatory process with its underhanded practices. The 10 CFR 50.69 proposed rulemaking was developed via an open, public multi-year process – at least, until NRC senior management started playing its games. UCS was extremely troubled to learn that the language in the proposed rulemaking that was issued by NRC for public comment differed significantly from that language developed through the open, public consensus process:

> ***"Nevertheless, the staff developed a draft version of the proposed rule which all internal stakeholders found to be acceptable (August 2, 2002, NRC external website version). Then, during the concurrence process, senior management made significant technical and policy adjustments to the proposed rule without providing a technical basis for the changes and without receiving an formal comments from stakeholders."*** [4]
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> **and**
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> ***"The staff in NRR has spent over two years developing the 50.69 rule language. This effort included numerous internal staff meetings, review by internal oversight groups, and public meetings with external stakeholders. This effort resulted in the July 31, 2002, version of the rule published on the NRC webs site (posted on August 2,). The July 321 version of the rule represented the balance of categorization and treatment requirements necessary to achieve a staff consensus to go forward with the proposed rulemaking. The Division of Regulatory Improvement Programs significantly altered the July 31 version of the rule without any input from the technical reviewers that were involved in the development of the rule for the past two years. Critical portions of the treatment process were eliminated based on the nebulous assertion that the rule language contained too much detail. The accompanying statement of considerations (SOC) indicates that the Commission expects licensees and applicants to satisfy many of the treatment provisions that were eliminated from the July 31 rule language. The current rule language is not consistent with many of the SOC expectations. As discussed in the ensuing paragraphs [deleted here], portions of the July 31 rule language were eliminated without a valid technical justification."*** [5]
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> **and**
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> ***"On July 31, 2002, the staff prepared a draft rule for Commission review that specified high-level requirements to provide sufficient regulatory treatment for plant SSCs consistent with the Commission papers describing the Option 2 effort. However, the 50.69 rulemaking package was significantly modified during the concurrence process. Based on my experience in component engineering and lessons learned from the Option 2 proof-of-concept effort, I consider the rulemaking package for proposed 10 CFR 50.69 submitted for Commission approval to be insufficient to maintain adequate protection of the public health and safety during operation of nuclear power plants implementing the rule."*** [6]

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[4] NRC memo dated September 26, 2002, from David C. Fischer, Senior Mechanical Engineer, to Samuel J. Collins, Director – Office of Nuclear Reactor Regulation, "Differing Professional View – Risk-Informed Part 50, Option 2," page 5.

[5] NRC memo dated September 26, 2002, from John R. Fair, Senior Mechanical Engineer, to Samuel J. Collins, Director – Office of Nuclear Reactor Regulation, "Differing Professional View Concerning the Proposed 10 CFR 50.69 Rulemaking," page 1.

[6] NRC memo dated September 26, 2002, from Thomas G. Scarbrough, Mechanical and Civil Engineering Branch, to Samuel J. Collins, Director – Office of Nuclear Reactor Regulation, "Differing Professional View Regarding Proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,"" page 2.

The NRC purports to adhere to the following "principles of good regulation":[7]

Independence **Nothing but the highest possible standards of ethical performance and professionalism should influence regulation. However, independence does not imply isolation. All available facts and opinions must be sought openly from licensees and other interested members of the public. The many and possibly conflicting public interests involved must be considered. Final decisions must be based on objective, unbiased assessments of all information, and must be documented with reasons explicitly stated.**

Openness **Nuclear regulation is the public's business, and it must be transacted publicly and candidly. The public must be informed about and have the opportunity to participate in the regulatory processes as required by law. Open channels of communication must be maintained with Congress, other government agencies, licensees, and the public, as well as with the international nuclear community.**

Efficiency **The American taxpayer, the rate-paying consumer, and licensees are all entitled to the best possible management and administration of regulatory activities. The highest technical and managerial competence is required, and must be a constant agency goal. NRC must establish means to evaluate and continually upgrade its regulatory capabilities. Regulatory activities should be consistent with the degree of risk reduction they achieve. Where several effective alternatives are available, the option which minimizes the use of resources should be adopted. Regulatory decisions should be made without undue delay.**

Clarity **Regulations should be coherent, logical, and practical. There should be a clear nexus between regulations and agency goals and objectives whether explicitly or implicitly stated. Agency positions should be readily understood and easily applied.**

Reliability **Regulations should be based on the best available knowledge from research and operational experience. Systems interactions, technological uncertainties, and the diversity of licensees and regulatory activities must all be taken into account so that risks are maintained at an acceptably low level. Once established, regulation should be perceived to be reliable and not unjustifiably in a state of transition. Regulatory actions should always be fully consistent with written regulations and should be promptly, fairly, and decisively administered so as to lend stability to the nuclear operational and planning processes.**

It's not apparent how senior management's shenanigans comports with these stated principles. For example, the Clarity principle states "*Agency positions should be readily understood and easily applied.*" As documented by the three DPV filers, the agency's rationale for excising important parts of the proposed rulemaking is not readily understood. UCS does not understand why the issued language deviated so much from the consensus language. How about the Openness principle that "*Nuclear regulation is the public's business, and it must be transacted publicly and candidly.*" What is the basis for NRC senior management setting aside two years of open, public work to derive a secret rulemaking language? Did NRC senior management acquiesce to NEI influence? Where was NRC senior management during the years it took to development the consensus language? Why didn't NRC senior management partake of the numerous public meetings and internal meetings during those years to steer the language to per its desires so external and internal stakeholders could understand?

Having principles is good. Following them is better. The NRC senior management desperately needs principles of better regulation.

[7] http://www.nrc.gov/who-we-are/values.html

The NRC must re-issue the proposed rulemaking with the basis for the language clearly articulated and available or revise its principles to match its practices. The language must be consistent with the statements of consideration and elements of the rulemaking package.

NRC senior management antics are sending a very strong message: it's pointless for the NRC staff and external stakeholders to participate in meetings to develop proposed rules. Instead, NRC staff and external stakeholders should simply wait for NRC management to secretly conjure up their version of the rule and comment on it. Message received. The NRC has a goal of "improving public confidence." Please put UCS down for "NO" with regard to meeting this goal.

4) In the proposed rule, the Commission is proposing to review and approve the categorization process to be used by the licensee. For treatment requirements, the proposed rule sets forth high-level requirements, and does not require NRC review and approval of specific processes a licensee would implement to meet these requirements. Another way to structure the rule would be to require NRC review and approval of the licensee's proposed treatment program for RISC-3 SSCs. The Commission is interested in any benefits of this approach as well as any implications for this rulemaking and its associated guidance.VI.2.3 Inspection and Enforcement.

UCS Response: It would be woefully inadequate for the NRC to merely review and approve the categorization process contemplated by the licensee seeking harbor under 10 CFR 50.69. Has the agency already forgotten its lessons learned from Davis-Besse? UCS calls attention to Lessons Learned Task Force recommendation 3.2.2(1):

> ***"The NRC should inspect the adequacy of PWR plant boric acid corrosion control programs, including their implementation effectiveness, to determine their acceptability for the identification of boric acid leakage, and their acceptability to ensure that adequate evaluations are performed for identified boric acid leaks."*** [8]

Davis-Besse demonstrated beyond a shadow of a doubt that the NRC must not rely solely on its approval of high-level criteria to ensure adequate public protection. The NRC accepted the high-level principles espoused in the B&W Owners Group boric acid corrosion control topical report. The NRC failed to verify that Davis-Besse was properly adhering to those accepted standards. It takes both a good process and good implementation for public safety to be assured.

In "special treatment" space, the high-level categorization process adopted by licensees will probably be based largely on the guidance[9] put out by the Nuclear Energy Institute (NEI). The NRC must go beyond virtual "spell checking" of each licensee's translation of the NEI guidance to the review and approval of specific processes a licensee would implement to meet the high-level criteria. The NRC's review must include an assessment of the metrics proposed by the licensee to monitor its reclassification efforts and ensure that components are being properly binned and that subsequent operating experience continues to warrant that binning. It is essential that NRC evaluate these metrics, not only to verify that the licensee's program contains self-checking but also to assist in the development and implementation of the NRC's module for post-reclassification inspections.

A report prepared by the Idaho National Engineering and Environmental Laboratory for the NRC established both the reason for and need for more than a review and approval of high-level criteria:

> ***"Plant processes will have a significant effect on providing reasonable confidence of component functionality, but the adequacy of the commercial standards and reduced plant processes would have to be evaluated on a plant-by-plant basis."*** [10]

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[8] NRC memo dated November 26, 2002, from Carl J. Paperiello to William D. Travers, "Senior Management Review of the Lessons-Learned Report for the Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head," attachment 1 page 3.

[9] Nuclear Energy Institute, NEI 00-04 Rev. C, "10 CFR 50.69 SSC Categorization Guideline," June 2002.

[10] J. H. Phillips, J. L. Edson, M. R. Holbrook, M. E. Nitzel, and A. G. Ware, Idaho National Engineering and Environmental Laboratory, NUREG/CR-6752, "A Comparative Analysis of Special Treatment Requirements for Systems, Structures, and Components (SSCs) of Nuclear Power Plants with Commercial Requirements of Non-Nuclear Power Plants," January 2002, page 56.

The NRC's review and approval must be deep enough to provide reasonable confidence of component functionality. The need for the NRC to do more than a superficial, high-level process review is also demonstrated by the enforcement action taken by the agency against the owner of Three Mile Island Unit 1.[11] The nearly quarter million dollar fine was levied by NRC because an NRC inspection team identified:

> ***"(1) inadequate engineering design controls, including incorrect design inputs for certain design basis calculations, inadequate verifications to assure the adequacy of design, and inadequate safety evaluations prior to making design changes; (2) poor implementation of the process for classifying components, resulting in a number of nuclear safety related components being downgraded to a lower classification without an appropriate safety evaluation or other supporting engineering documentation; (3) failure to ensure the RB [reactor building] emergency cooling fans were environmentally qualified; (4) failure to take timely and appropriate corrective actions for conditions adverse to quality that existed at the facility, including conditions related to the Decay Heat Removal system, to the quality assurance findings regarding inappropriate equipment classification downgrades, and to the environmental qualification deficiency".***

All four of these violations are pertinent to the 10 CFR 50.69 rulemaking. The first violation relates to the PRA scope and quality issue. The NRC sanctioned TMI's owners for making decisions based on an inadequate design bases foundation. That very same issue certainly applies to decisions made on limited scope PRAs of uncertain quality. The second violation relates to the implementation of the high-level guidance of 10 CFR 50.69. The NRC sanctioned TMI's owners not for having an inadequate process but for inadequate implementation of an acceptable process. Likewise, the NRC needs to assess both the 10 CFR 50.69 reclassification process AND ITS IMPLEMENTATION. The third violation relates to verifying that components placed into RISC bins conform with the established criteria for those bins. The NRC sanctioned TMI's owner because it failed to meet ALL of the required attributes for a component's classification. The fourth violation relates to the metrics used to ensure that RISC binning is proper and any binning errors are corrected. The NRC sanctioned TMI's owners because it failed to remedy classification mistakes in a timely manner after they were identified.

The NRC must take the same approach for 10 CFR 50.69 as it did at TMI: (1) verify that plant owners are using full-scope, high quality PRAs, (2) verify that plant owners not only have adequate high-level process guidance, but are also adequately implementing their processes, (3) verify that components conform with all the established criteria for placement in RISC bins, and (4) verify that any RISC binning errors are found and corrected in a timely manner.

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[11] Letter dated October 8, 1997, from Hubert J. Miller, Regional Administrator, Nuclear Regulatory Commission, to James W. Langenbach, Vice President and Director – TMI, GPU Nuclear Corporation, "Notice of Violation and Proposed Imposition of Civil Penalties - $210,000 (NRC Inspection Reports Nos. 50-289/96-201; 97-01; 97-02; 97-03; & 97-04)."

5) The Commission recognizes that the final rule may have implications with respect to NRC's reactor oversight process including the inspection program, and enforcement. In its final decision on this rulemaking, the Commission proposes to document its conclusions as to whether or not new or revised inspection or enforcement guidance is necessary. Public comment is requested on whether or not changes are needed in our inspection and enforcement programs to enable NRC to exercise the appropriate degree of regulatory oversight of these aspects of the facility operation. VI.2.4 Operating Experience.

UCS Response: Because the proposed rule is voluntary, some plant owners may adopt it while others may opt to stay with the existing regulations. The plants owners who pursue it may do so at their convenience. The NRC's oversight capability may be diminished by having some owners reclassifying components under 10 CFR 50.69 and other owners not doing so. Consider the hypothetical situation where the owner of one facility with a single unit (Reactor U) is approved by the NRC for 10 CFR 50.69 treatment while the owner of another virtually identical facility with two units (Reactor W) sticks with the existing regulations. If a widget at Reactor U that was formerly considered safety-related but is now classified RISC-3 is found to be degraded, the component's reclassification may have moved this discovery below the reporting thresholds of 10 CFR 21 and 10 CFR 50.72/50/73. Conversely, impairment of the same widget on one of the units at Reactor W might result in a report to the NRC.

Inconsistent reporting of identical component failures causes problems for the NRC on both sides of the inconsistency. When the reclassification drops component failures below the reporting threshold, it prevents the NRC and other plant owners from knowing about the problem and taking appropriate steps to fix it elsewhere. When a component failure is reported by Reactor W, it complicates NRC's responses in ways such as: (a) determining addresses for generic communications (e.g., should the owner of Reactor U get the correspondence or not), (b) communicating to the public why a reportable event at Reactor W is not even an inspectable area at Reactor U, (c) making it extremely difficult to take enforcement action against the Reactor W for inadequate corrective actions on a broken widget when that same widget on Reactor U is not even inspectable, and (d) creating non-conservative databases of component failures.

The NRC has issued numerous generic communications regarding impaired components, including the following abridged listing:

- **Information Notice 88-19, "Questionable Certification of Class 1E Components," dated April 26, 1988: The owner of the Wolf Creek nuclear plant purchased 60 fuses for Class 1E electrical components that were supposed to meet specifications for materials and environmental qualifications. Upon receipt, the Quality Department at Wolf Creek determined that the fuses were substandard.**
- **Information Notice 81-08, "Repetitive Failures of Limitorque Operator SMB-4 Motor-to-Shaft Key," dated March 20, 1981: After the RHR "B" loop suppression pool cooling inboard throttle valve at the Cooper nuclear plant failed to operate, it was determined that the motor operator had the wrong steel key. Similar failures were subsequently discovered at the Pilgrim, Hatch, and FitzPatrick nuclear power plants.**
- **Information Notice 90-57, "Substandard, Refurbished Potter & Brumfield Relays Misrepresented as New," dated September 5, 1990: The owner of the Harris nuclear plant ordered 22 new relays for use in the emergency diesel generator safety bus sequencer circuits. The owner was suspicious when the relays arrived much sooner than the typical 10-12 week lead time. When the suspect relays were tested, all failed one or more of the performance tests. Disassembly of the relays revealed assortments of nonstandard, substandard, and obsolete parts in incorrect configurations.**

- **Information Notice 83-84, "Cracked and Broken Piston Rods in Brown Bovery Electric Type 5HK Breakers," dated December 30, 1983: The owner of the Fermi nuclear plant informed the NRC about a problem with parts within some electrical breakers used at the site. The substandard parts impaired "breaker's capability to interrupt at low current" and shortened the maintenance interval. Breakers of the same manufacturing batch were subsequently determined to have been shipped to 38 operating reactors in the US and abroad.**
- **Information Notice 88-97, "Potentially Substandard Valve Replacement Parts," dated December 16, 1988: The owner of the Palisades nuclear plant sent a turbine bypass valve to an outside vendor for refurbishment. The vendor also received a package of authorized replacement valve parts from Palisades. The refurbished bypass valve was reinstalled and the plant restarted. Then the owner learned that the owner had not used the authorized replacement parts. The safety function of the bypass valve with substandard parts could not be assured.**
- **Information Notice 81-28, "Failure of Rockwell-Edward Main Steam Isolation Valves," dated September 3, 1981: Several failures of main steam isolation valves caused by separation of the valve disc from the valve stem were reported by the owners of the Hatch and Brunswick nuclear plants. The failures occurred at threaded connections and apparently resulted from (a) improper manufacturing tolerances and (b) excessive vibrations during operation damaging threads.**
- **Information Notice 86-66, "Potential for Failure of Replacement AC Coils Supplied by the Westinghouse Electric Corporation for Use in Class 1E Motor Starters and Contactors," dated August 15, 1986: The NRC received a Part 21 report from Westinghouse about higher-than-normal failures rates from ac coils manufactured at their Puerto Rican facility. The ac coil failures occurred within a few hours after being energized.**
- **Information Notice 88-46, "Licensee Report of Defective Refurbished Circuit Breakers," dated July 8, 1988: The owner of the Diablo Canyon nuclear plant ordered 30 new non-safety-related electric circuit breakers manufactured by the Square D company. Investigation of the 30 "new" circuit breakers revealed that they were refurbished parts being passed off as new. None of the "new" breakers passed Square D and Underwriters Laboratory tests.**
- **Information Notice 81-06, "Failure of ITE Model K-600 Circuit Breaker," dated March 11, 1981: The owner of the Rancho Seco nuclear plant (now defunct) informed the NRC that an electrical circuit breaker failed to trip. Subsequent investigation revealed that the tripping coil wire was improperly sized and had slipped out of its terminal causing the failure. An extent-of-condition assessment found that ITE Model K-1600 breakers may also have improperly sized tripping coil wires.**
- **Information Notice 86-81, "Broken Inner-External Closure Springs on Atwood & Morrill Main Steam Isolation Valves," dated September 15, 1986: The owner of the Fermi Unit 2 reactor informed the NRC that it had discovered broken springs inside main steam isolation valves. The springs failed due to quench cracking caused by the heat treatment process during manufacturing. The broken springs could cause the main steam isolation valves to close slower than the 3 to 5 seconds specified in the operating license and to fail to seal tightly against leakage.**
- **Information Notice 82-04, "Potential Deficiency of Certain Agastat E-7000 Series Time-Delay Relays," dated March 10, 1982: A vendor submitted a Part 21 report to the NRC regarding the recall of all relays it manufactured between July 15, 1981 and January 12, 1982, because of a defect. At elevated temperature within the required operating range of the relays, fluid can leak from a pneumatic timing diaphragm to cause the relays to actuate at less than the specified time delay.**
- **Information Notice 90-65, "Recent Orifice Plate Problems," dated October 5, 1990: The owner of the San Onofre nuclear plant informed the NRC that the indicated feedwater flow had been lower than actual feedwater flow due to an orifice in the flow indicator**

being installed backwards. The reactor's actual power level may have been about 4 percent higher than calculated due to the flow indication error. Flow orifices were also found to be installed backwards at the Farley, Harris, Salem, Brunswick, Waterford, North Anna, and Surry nuclear power plants.

The post-event inquiries into the March 1979 meltdown of the Three Mile Island Unit 2 reactor criticized the NRC for having known about a precursor event at another nuclear facility but not warning other plant owners about the known vulnerability:

> ***"On September 24, 1977, there was a minor accident at the Davis-Besse Unit 1 plant operated by Toledo Edison in Ohio. It was also very similar to the early minutes of the Three Mile Island accident. ... Toledo Edisin, Babcock & Wilcox, who had supplied the reactor, and the NRC all analyzed the accident. ... However, according to the NRC, "no significant action resulted from this effort." ... Some months later, James Creswell, a Region III inspector, did raise several issues. One was that the operators might have incorrectly turned off HPI. As a result, emergency procedures at Davis-Besse were modified to caution operators against turning off HPI in the event of a leak in the pressurizer. This possibility was not recognized as a generic safety concern, and the NRC failed to take further action or to notify other utilities."***[12]

The 10 CFR 50.69 rulemaking provides an alternate way for the NRC to repeat this tragic mistake. For example, suppose that defective fuses are discovered at a reactor that has adopted 10 CFR 50.69 and that these fuses are for Class 1E components that have been reclassified as RISC-3. The NRC might not be informed of the degraded fuses because they fall below the reporting threshold. Thus, the NRC will not be able to warn other plant owners of the problem as it did with Information Notice 88-19 above. Or suppose that defective parts are found in electrical breakers as described in Information Notice 83-84 above. If the defective parts are within electrical breakers for RISC-3 components, the NRC may not be informed about it. Thus, the 10 CFR 50.69 rulemaking can return the NRC to its pre-TMI dark ages.

[12] United States Senate Committee on Environment & Public Works, Subcommittee on Nuclear Regulation, "Nuclear Accident and Recovery at Three Mile Island: A Special Investigation," June 1980, pp. 79-80.

6) One of the areas of uncertainty associated with this rulemaking has been the potential effects of changes in treatment on SSC reliability and common-cause failure potential. This is reflected in the requirement for evaluations (sensitivity studies) to provide reasonable confidence that any potential increase in risk would be small, with a basis provided for the factors to be assumed in these evaluations. Further, the rule requires the licensee to consider performance information to determine whether there are any adverse changes such that SSC unreliability values approach the values used in these evaluations, and to make necessary adjustments to the categorization and treatment processes. As discussed in Section VII.2, below, draft RG (DG-1121) provides some discussion about techniques that might be used in determining the factors for these evaluations.

UCS Response: The proposed rulemaking would allow plant owners to reclassify certain equipment from safety-related, important-to-safety, and so on to "low safety significant" equipment (i.e., RISC-3 components). The reclassified RISC-3 components would no longer be subjected to the high quality standards afforded safety-related equipment by federal regulations. Instead, the RISC-3 components would be governed by commercial codes and standards. [Indeed, it is the lower cost associated with equipment purchased and maintained to these lower standards that is the engine driving this rulemaking.] A report by the Idaho National Engineering and Environmental Laboratory for the NRC concluded that lowering the standards compromises safety margins unless precautions are taken:

> ***"Discussions with utility representatives ... lead to the conclusion that commercial codes and standards alone do not provide the processes necessary to provide reasonable confidence of functionality. ... The overall conclusions ... are that commercial codes and standards by themselves are insufficient to provide reasonable confidence of SSC functionality. However, the critical attributes missing in commercial codes and standards could be supplied by (1) measures such as utilization of detailed engineering specifications, (2) plant processes and procedures, (3) multilevel QA programs that provide less rigor than 10 CFR 50, Appendix B, but augment commercial requirements, or (4) a combination of these approaches. Therefore, for the NRC to allow the nuclear industry to use commercial practices for procurements of replacement RISC-3 SSCs [systems, structures, and components], they would have to rely heavily on the good judgment and internal processes of the nuclear plants, realizing that there may be minimal documentation or in-service test/inspection results to give reasonable confidence of functionality."***[13]

The NRC' generic communications contain numerous examples of substandard components being used at US nuclear power plants. Examples include:

- **Information Notice 88-19, "Questionable Certification of Class 1E Components," dated April 26, 1988: The owner of the Wolf Creek nuclear plant purchased 60 fuses for Class 1E electrical components that were supposed to meet specifications for materials and environmental qualifications. Upon receipt, the Quality Department at Wolf Creek determined that the fuses were substandard.**
- **Information Notice 90-57, "Substandard, Refurbished Potter & Brumfield Relays Misrepresented as New," dated September 5, 1990: The owner of the Harris nuclear plant ordered 22 new relays for use in the emergency diesel generator safety bus sequencer circuits. The owner was suspicious when the relays arrived much sooner than the typical 10-12 week lead time. When the suspect relays were tested, all failed one or**

[13] J. H. Phillips, J. L. Edson, M. R. Holbrook, M. E. Nitzel, and A. G. Ware, Idaho National Engineering and Environmental Laboratory, NUREG/CR-6752, "A Comparative Analysis of Special Treatment Requirements for Systems, Structures, and Components (SSCs) of Nuclear Power Plants with Commercial Requirements of Non-Nuclear Power Plants," January 2002, page 53.

more of the performance tests. Disassembly of the relays revealed assortments of nonstandard, substandard, and obsolete parts in incorrect configurations.

- Information Notice 83-84, "Cracked and Broken Piston Rods in Brown Bovery Electric Type 5HK Breakers," dated December 30, 1983: The owner of the Fermi nuclear plant informed the NRC about a problem with parts within some electrical breakers used at the site. The substandard parts impaired "breaker's capability to interrupt at low current" and shortened the maintenance interval. Breakers of the same manufacturing batch were subsequently determined to have been shipped to 38 operating reactors in the US and abroad.
- Information Notice 88-97, "Potentially Substandard Valve Replacement Parts," dated December 16, 1988: The owner of the Palisades nuclear plant sent a turbine bypass valve to an outside vendor for refurbishment. The vendor also received a package of authorized replacement valve parts from Palisades. The refurbished bypass valve was reinstalled and the plant restarted. Then the owner learned that the owner had not used the authorized replacement parts. The safety function of the bypass valve with substandard parts could not be assured.
- Information Notice 86-66, "Potential for Failure of Replacement AC Coils Supplied by the Westinghouse Electric Corporation for Use in Class 1E Motor Starters and Contactors," dated August 15, 1986: The NRC received a Part 21 report from Westinghouse about higher-than-normal failures rates from ac coils manufactured at their Puerto Rican facility. The ac coil failures occurred within a few hours after being energized.
- Information Notice 88-46, "Licensee Report of Defective Refurbished Circuit Breakers," dated July 8, 1988: The owner of the Diablo Canyon nuclear plant ordered 30 new non-safety-related electric circuit breakers manufactured by the Square D company. Investigation of the 30 "new" circuit breakers revealed that they were refurbished parts being passed off as new. None of the "new" breakers passed Square D and Underwriters Laboratory tests.
- Information Notice 81-06, "Failure of ITE Model K-600 Circuit Breaker," dated March 11, 1981: The owner of the Rancho Seco nuclear plant (now defunct) informed the NRC that an electrical circuit breaker failed to trip. Subsequent investigation revealed that the tripping coil wire was improperly sized and had slipped out of its terminal causing the failure. An extent-of-condition assessment found that ITE Model K-1600 breakers may also have improperly sized tripping coil wires.
- Information Notice 86-81, "Broken Inner-External Closure Springs on Atwood & Morrill Main Steam Isolation Valves," dated September 15, 1986: The owner of the Fermi Unit 2 reactor informed the NRC that it had discovered broken springs inside main steam isolation valves. The springs failed due to quench cracking caused by the heat treatment process during manufacturing. The broken springs could cause the main steam isolation valves to close slower than the 3 to 5 seconds specified in the operating license and to fail to seal tightly against leakage.
- Information Notice 82-04, "Potential Deficiency of Certain Agastat E-7000 Series Time-Delay Relays," dated March 10, 1982: A vendor submitted a Part 21 report to the NRC regarding the recall of all relays it manufactured between July 15, 1981 and January 12, 1982, because of a defect. At elevated temperature within the required operating range of the relays, fluid can leak from a pneumatic timing diaphragm to cause the relays to actuate at less than the specified time delay.

These substandard and defective parts were discovered due to the high level of receipt inspection and ongoing preventative maintenance required for safety-related equipment – standards that are lower for components reclassified as RISC-3. The higher standards will be retained for the RISC-1 and RISC-2 components. But those higher standards did not prevent many of the components described in the Information Notices listed above from being installed and used for awhile in their defective condition. The higher standards did provide more

opportunities for detection of substandard parts and also provided for wider dissemination of information about the substandard parts. The lower standards for components reclassified as RISC-3 makes it more likely that nuclear power plants will operate with substandard parts, thus increasing the potential for common-mode failures.

The NRC has every right to be concerned about common-cause failure potential from reclassified equipment. The PRAs that will sort equipment into the four RISC bins primarily determine how core damage frequency (CDF) or large, early release frequency (LERF) is affected by the postulated failure of the equipment to function (i.e., active failures). In other words, what is the impact of an isolation valve failing to close upon demand or of a makeup water pump failing to run when needed? The PRAs do not, except in rare cases, evaluate the potential for equipment failure to prevent other equipment from functioning (i.e., passive failures leading to active failures) or to increase the potential for operator error. The classic example is probably the Indian Point Unit 2 event where approximately 100,000 gallons of water leaked from service water lines into containment over time and flooded the reactor cavity to the point where several feet of the reactor pressure vessel was submerged.[14] Another classic example would be the power supply failure for non-nuclear instrumentation at the Crystal River nuclear plant. The seemingly innocuous failure of a 24 volt dc power supply caused a power operated relief valve on the pressurizer to stay open and control room indication of safety system equipment to be lost. The operator conservatively decided to keep the high pressure injection pump running since he/she was unable to monitor 70% of the instrumentation normally available. As a result, the pressurizer was pumped solid and the overflow blew the rupture disc on the reactor coolant drain tank, flooding the containment with about 43,000 gallons of reactor water.[15] While the operator erred on the safe side in this event, there was the very real potential for a different "guess" to rob the reactor core of essential cooling water.

According to former NRC Chairman Meserve: "*There is the potential for the common-cause failure of RISC-3 SSCs [systems, structures, and components] under accident conditions.*"[16] The rule, in its present incarnation, fails to compensate for the increased risk of common-mode failures. Thus, safety margins will be compromised by this rule.

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[14] Nuclear Regulatory Commission, Information Notice 80-37, "Containment Cooler Leaks and Reactor Cavity Flooding at Indian Point Unit 2," October 24, 1980.
[15] Nuclear Regulatory Commission, Information Notice 80-10, "Partial Loss of Non-Nuclear Instrument System Power Supply During Operation," March 7, 1980.
[16] Nuclear Regulation Commission, Commission Voting Record for SECY-02-0176, March 28, 2003.

7) The Commission is interested in the role that relevant operational experience could play in reducing the uncertainty associated with the effects of treatment on performance and specifically seeks public comment as to what information might be available and how it could be used to support implementation of this rulemaking.

UCS Response: A primary objective of this proposed rulemaking is to move "special treatment" equipment having low safety significance from NRC control to owner control. Relevant operational experience strongly suggests that the Commission should seek the opposite objective of reducing the inventory of equipment under owner control. When the March 22, 1975, fire at the Browns Ferry Nuclear Plant in Alabama disabled all of the emergency core cooling systems on Unit 1, it was the non-safety-related control rod drive system pump that saved the day by continuing to supply vitally needed cooling water to the reactor vessel. When a pineapple-sized hole was discovered in the safety-related, ASME code reactor vessel at Davis-Besse, it was the non-safety-related and non-ASME code stainless steel liner that prevented a disastrous loss of coolant accident. There are many, many other examples of industry and NRC discounting the failure of safety-related equipment because non-safety-related equipment remained available to carry on. Thus, rather than reduce regulatory oversight of equipment that is so often credited with lowering risk, the prudent thing to do would be to increase oversight. The least imprudent thing to do would be to retain the status quo.

"Relevant operating experience" also argues <u>against</u> this rulemaking in other important ways. Plant owners have been required to report impairments of safety-related equipment under 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73, "License event report system." These regulations do not require plant owners to report impairments of non-safety-related equipment, except under limited circumstances. The nuclear industry voluntarily, and essentially "secretly," maintains data on impairments of safety-related and non-safety-related equipment via the Nuclear Plant Reliability Data System (NPRDS) and its successor the Equipment Performance Information Exchange (EPIX). This background is the basis for the following concerns about "relevant operating experience":

1. **The NRC has issued considerable guidance to help plant owners figure out how to report impairments of safety-related equipment. For example, Revision 2 to NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," was issued in October 2000 to supplement the guidance provided in Revisions 0 and 1 of this document. And the NRC issued Regulatory Issue Summary 2001-014, "Position on Reportability Requirements for Reactor Core Isolation Cooling System Failure," on July 19, 2001. The ever-growing list of Frequently Asked Questions (FAQs) about reporting safety system unavailability under the Reactor Oversight Process (ROP) attests to ongoing confusion. FAQs 289, 322, 325 and 335 involve unavailability of emergency diesel generators – safety-related components that have the same design and function as twenty years ago but remain cloaked in confusion over what degree of impairment constitutes unavailability. These documents, and many more like them, demonstrate beyond reasonable doubt that thresholds for reporting impairments of safety-related equipment have long been and continue to remain confusing. The chronic inability to establish clean criteria for consistently reporting failures of safety-related equipment strongly suggests that clean criteria have not been developed for reporting failures of non-safety-related equipment to NPRDS and/or EPIX.**
2. **The public has access to impairments reported by plant owners to the NRC under 10 CFR 50.72 and 50.73 (but not, obviously, impairments <u>not</u> reported by plant owners due to the "fog of failure"). The public does not have access to impairments reported by plants owners to NPRDS and now EPIX. The NRC itself has very limited access to impairments reported to NPRDS and EPIX. If 10 CFR 50.69 is adopted as currently proposed, equipment impairments that had been reported to NRC will no longer be**

reported to NRC. Neither the NRC nor the public will be able to access NPRDS/EPIX to see if equipment failure rates are rising. Based on the South Texas Project pilot effort, equipment that would be "hidden" from both NRC and the public includes:[17]

a. **Emergency diesel generator air start valves**
b. **Main steam isolation valves**
c. **Spent fuel pool pumps and valves**
d. **Most residual heat removal (RHR) system valves**
e. **All but one valve in the service water system**
f. **Motor-operated valves in both the high pressure safety injection (HPSI) and low pressure safety injection (LPSI) systems**
g. **Motor-operated valves in the component cooling water system**
h. **Containment spray pumps and valves**
i. **Most containment isolation valves**

Thus, if the reclassification resulted in the unavailability of this equipment approaching unity (in other words, it's broken almost all the time), neither the NRC nor the public would have any clue – until it was too late.

It is also contradictory for the NRC to facilitate downgrading the regulatory significance of the containment spray pumps and valves and the HPSI and LPSI systems at the same time it is upgrading the regulatory significance of the containment recirculation sump screens (i.e., Bulletin 2003-01 dated June 9, 2003). The NRC's sudden interest in containment sump functioning and resolution of GSI-191 demonstrates that this equipment plays an important role in protecting public health and safety. The NRC would compromise public health and safety if it reduces regulatory oversight of such equipment.

Responses prepared by: David Lochbaum, Nuclear Safety Engineer, UCS

[17] NRC memo dated September 26, 2002, from David C. Fischer to Samuel J. Collins, "Differing Professional View – Risk-Informed Part 50, Option 2," (ADAMS Accession No. ML022690452), page 2.

Before the
UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISISON

In the matter of **Proposed Rule on Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors**	**10 CFR 50** **RIN 3150-AG42**

DECLARATION OF DAVID A. LOCHBAUM, NUCLEAR SAFETY ENGINEER, UNION OF CONCERNED SCIENTISTS, CONCERNING PROPOSED RULE ON RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS

I, David A. Lochbaum, make the following declaration:

1. My name is David A. Lochbaum. I reside in the state of Maryland.

2. I am employed by the Union of Concerned Scientists as their nuclear safety engineer. I have been so employed since October 1996. The Union of Concerned Scientists, with offices located at 1707 H Street NW Suite 600, Washington, DC 20006, is an independent, nonprofit organization dedicated to advancing responsible public policies in areas where technology plays a critical role.

3. I have the following responsibilities at UCS: a) directing and coordinating the nuclear safety project; b) monitoring developments in the nuclear industry for potential impact on safety margins; c) serving as technical authority and spokesperson on nuclear issues; and d) initiating legal action to correct safety problems.

4. I have worked in the field of nuclear engineering since June 1979. I am a graduate of the University of Tennessee with a bachelor of science in nuclear engineering.

5. After receiving my nuclear engineering degree, I went to work for the Georgia Power Company as a junior engineer at their Edwin I. Hatch Nuclear Power Plant. I held various positions in the commercial nuclear power industry over the next 17 years prior to joining UCS. This experience is detailed in the resume attached hereto as Exhibit A.

6. I am the author of *Nuclear Waste Disposal Crisis* (Pennwell Books, Tulsa, January 1996) on the technical problems with spent fuel storage at reactor sites and numerous reports for UCS on nuclear safety issues.

7. I have reviewed the *Federal Register* notice dated May 16, 2003, (Vol. 68, No. 95) for the proposed rule along with its supporting documentation. I have also examined and am familiar with, for the purposes of preparing this declaration, the applicable federal regulations contained in Title 10 of the Code of Federal Regulations. I have relied upon these documents in formulating my opinions as expressed in this declaration.

8. Having examined the relevant documents as mentioned above, it is my professional opinion that the proposed rule, if adopted in its present form, undermines nuclear safety margins and makes it more likely that nuclear plant workers and members of the general public will be harmed by radiation from nuclear power plants for the following reasons:

(a) The proposed rule would allow nuclear plant owners to reclassify safety-related equipment to risk-informed safety class 3 (RISC-3) if the equipment is determined to have low safety significance based on probabilistic risk assessment (PRA) results and/or an expert panel. As detailed further in Exhibit B, the rule fails to either (a) require that the PRA used in these determinations be all encompassing (i.e., all modes of operation and for both internal and external events) or to meet or exceed minimum quality standards or (b) require that the equipment reclassified to RISC-3 status be confined to only that equipment having no safety functions other than at-power (i.e., within the narrow scope of the PRA). As a result, equipment with safety functions may be mistakenly reclassified as RISC-3. As RISC-3 equipment, they will be subject to less stringent procurement and surveillance standards. Thus, the reliability of equipment with safety functions is likely to be reduced by this rule and the threat to workers and people living near nuclear power plants adopting this rule will increase.

(b) As detailed in Exhibit B, the proposed rule would essentially allow plant owners to downgrade safety-related equipment based on results from PRAs and/or expert panels showing that the equipment has low safety significance. Yet it is very possible that the reason the equipment demonstrates "low safety significance" is that its failure rates are low (i.e., it is not causing many near-misses at the plants). It is likely that a major reason for the low failures rates are the high standards applied to the procurement and operational testing of safety-related equipment. But if the equipment is reclassified to RISC-3, the procurement and operational testing standards are relaxed. The NRC's Reactor Oversight Process is unlikely to detect increased failure rates of RISC-3 equipment in a timely manner since NRC inspectors are rigidly trained to devote their energies to areas other than "low safety significance." Thus, it is likely that the rule will increase the risk from common-cause failures of equipment.

(c) The proposed rule will reduce the information reported to the NRC about substandard and malfunctioning equipment as explained further in Exhibit B. Under present regulations, the NRC is informed about safety-related equipment with problems

above a threshold. By reclassifying safety-related equipment to RISC-3, the NRC will likely receive fewer reports about problems affecting that equipment. This could adversely affect safety levels because while Plant X might very well be using Widget A in a low safety significant manner, an identical widget may be used at Plants Y and Z in a higher safety significant manner. If the NRC no longer hears about Widget A problems from Plant X, it can no longer warn Plants Y and Z about them. Thus, the rule is likely to result in more cases like the Davis-Besse/Three Mile Island Unit 2 case. In September 1977, Davis-Besse experienced a problem at low power that included, among other things, a power-operated relief valve (PORV) pressurizer sticking open. The open PORV caused the level in the pressurizer to read higher than actual. Workers at Davis-Besse revised procedures about problems that can result from the pressurizer PORV being open. The NRC knew about this event and the steps taken to remedy it at Davis-Besse, but failed to warn other plant owners. In March 1979, Three Mile Island Unit 2 experienced a very similar problem, except that it began with the reactor at nearly full power. As at Davis-Besse, a PORV stuck open. Lacking the procedural guidance that Davis-Besse operators had, the TMI operators relied on the false pressurizer water level indication and took steps that had the unintended consequence of making matters worse. The rule will restrict the dissemination of problem information and increase the chances of nuclear plants operating with problems that have already been found and fixed elsewhere.

9. It is my professional opinion that the safety concerns addressed in paragraph 8 could be created by the promulgation of the rule in its present form. I am also of the professional opinion, and do so state here, that the risk to persons working at the plant and/or living in close proximity to the facility could be increased by promulgation of the proposed rule, and the risks and potential are real, not highly speculative, and should be taken very seriously.

I declare under penalty of perjury that the foregoing declaration is true and correct.

Executed July 16, 2003

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List of Exhibits:

A – Resume of David Lochbaum
B – Comments on Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors

David A. Lochbaum

Experience Summary

10/96 to date *Nuclear Safety Engineer, Union of Concerned Scientists*

Responsible for directing UCS's nuclear safety program, for monitoring developments in the nuclear industry, for serving as the organization's spokesperson on nuclear safety issues, and for initiating action to correct safety concerns.

11/87 to 09/96 *Senior Consultant, Enercon Services, Inc.*

Responsible for developing the conceptual design package for the alternate decay heat removal system, for closing out partially implemented modifications, reducing the backlog of engineering items, and providing training on design and licensing bases issues at the Perry Nuclear Power Plant.

Responsible for developing a topical report on the station blackout licensing bases for the Connecticut Yankee plant.

Responsible for vertical slice assessment of the spent fuel pit cooling system and for confirmation of licensing commitment implementation at the Salem Generating Station.

Responsible for developing the primary containment isolation devices design basis document, reviewing the emergency diesel generators design basis document, resolving design document open items, and updating design basis documents for the James A. FitzPatrick Nuclear Power Plant.

Responsible for the design review of balance of plant systems and generating engineering calculations to support the Power Uprate Program for the Susquehanna Steam Electric Station.

Responsible for developing the reactor engineer training program, revising reactor engineering technical and surveillance procedures and providing power manuevering recommendations at the Hope Creek Generating Station.

Responsible for supporting the lead BWR/6 Technical Specification Improvement Program and preparing licensing submittals for the Grand Gulf Nuclear Station.

03/87 to 08/87 *System Engineer, General Technical Services*

Responsible for reviewing the design of the condensate, feedwater and raw service systems for safe shutdown and restart capabilities for the Browns Ferry Nuclear Plant.

08/83 to 02/87 *Senior Engineer, Enercon Services, Inc.*

Responsible for performing startup and surveillance testing, developing core monitoring software, developing the reactor engineer training program, and supervising the reactor engineers and Shift Technical Advisors at the Grand Gulf Nuclear Station.

Experience Summary (continued)

10/81 to 08/83 *Reactor Engineer / Shift Technical Advisor, Tennessee Valley Authority*

Responsible for performing core management functions, administering the nuclear engineer training program, maintaining ASME Section XI program for the core spray and CRD systems, and covering STA shifts at the Browns Ferry Nuclear Plant.

06/81 to 10/81 *BWR Instructor, General Electric Company*

Responsible for developing administrative procedures for the Independent Safety Engineering Group (ISEG) at the Grand Gulf Nuclear Station.

01/80 to 06/81 *Reactor Engineer / Shift Technical Advisor, Tennessee Valley Authority*

Responsible for directing refueling floor activities, performing core management functions, maintaining ASME Section XI program for the RHR system, providing power maneuvering recommendations and covering STA shifts at the Browns Ferry Nuclear Plant.

06/79 to 12/79 *Junior Engineer, Georgia Power Company*

Responsible for completing pre-operational testing of the radwaste solidification systems and developing design change packages for modifications to the liquid radwaste systems at the Edwin I. Hatch Nuclear Plant.

Education

June 1979 Bachelor of Science in Nuclear Engineering, The University of Tennessee at Knoxville

May 1980 Certification, Interim Shift Technical Advisor, TVA Browns Ferry Nuclear Plant

April 1982 Certification, Shift Technical Advisor, TVA Browns Ferry Nuclear Plant

Professional Affiliations

Member, American Nuclear Society (since 1978).

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