UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MEMORANDUM FOR: NRR Division Directors
NRR Deputy Directors
NRR Assistant Directors
NRR Branch Chiefs
NRR Section Leaders

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: NRR MONTHLY TECHNICAL REPORT FOR APRIL 1982

Enclosed for your information is the NRR Monthly Technical Report for April 1982.

Inputs for the report of May 1982 are due by June 8.

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
NRR Monthly Technical Report

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MONTHLY TECHNICAL REPORT

OFFICE OF NUCLEAR REACTOR REGULATION

April 1982

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PSYCHOLOGICAL STRESS ASSOCIATED WITH RESTART OF TMI-1

Division of Engineering

Contact is Jan Norris, Siting Anlysis Branch

On January 7, 1982, the U. S. Court of Appeals for the D. C. Circuit directed the NRC to prepare an environmental assessment of the effects of the proposed restart of Three Mile Island Unit 1 on the psychological health of neighboring residents and on the well-being of the communities surrounding Three Mile Island. NRC staff is in the process of preparing such an assessment. To ascertain the state of knowledge of psychological stress, the NRC asked the MITRE Corp. to convene a workshop on February 4-5, which included eleven nationally recognized experts on the subject. The proceedings were published in April as NUREG/CP-0026.

Many participants stated that generalized predictions of stress responses associated with a restart of TMI-1 can be made, but the limitations of social-science theory and methodology, as well as inadequate data, are likely to yield predictions in which they would not place a high degree of confidence. Short-term or acute responses are more amenable to prediction than long-term or chronic responses. Stress responses to a TMI-1 restart are expected to be lower in magnitude than those associated with the TMI-2 accident, possibly being more comparable to those associated with the later venting of krypton gas from TMI-2.

A body of literature exists on responses to natural disasters and other transient stresses, but the issue of its applicability to a TMI-1 restart remained unresolved. Some participants argued in favor of extrapolating from this literature, contending that the qualitative similarities between the TMI-2 accident and natural disasters make such extrapolation appropriate. Others argued that the TMI-2 accident was too minor an event to define as a disaster and that extrapolation from disaster literature would overestimate the stress of a TMI-1 restart. Still others agreed that extrapolation from disaster literature is inappropriate, but based their position on the argument that, having been subjected to the TMI-2 accident and sensitized by the experience, the TMI population is unique.

The intangible nature of the consequences to the community resulting from the TMI-2 accident is a major characteristic differentiating the accident from a natural disaster. The lack of visible damage prevented many residents from clearly defining what had occurred and from taking corrective measures. The fact that a possible restart of TMI-1 involves anticipatory stress makes analogies to natural disasters additionally difficult.

The concept of nuclear power held by the public affects the psychological impacts of this technology. Studies referred to by one participant indicate a common belief that nuclear power is a risk or threat second in magnitude only to nuclear war. The public understanding of nuclear power is affected in a major way by the media, which provides the only contact of most individuals with the technology.

The expert participants recommended that existing TMI data be reanalyzed and that additional data be gathered about the TMI population to establish improved baselines, to identify high-risk groups, and to assess possible psychological stress impact of a decision not to restart TMI-1. Ameliorative actions that might lessen psychological stress impact of a TMI-1 restart should focus on education, information access, and counseling. Actions should be implemented only after the public is assured of the safety of TMI-1 and the case for restart is convincingly presented. The media, particularly interactive television, could contribute to keeping the public well informed.

COMPLIANCE WITH SHIFT MANNING REQUIREMENTS

Division of Human Factors Safety

By Clare Goodman and J. J. Persensky

In accordance with NUREG-0737, "Clarification of TMI Action Plan Requirements," there is to be a licensed senior reactor operator (SRO) in the control room of a nuclear power unit at all times other than during cold shutdown conditions. This is in addition to the shift supervisor, who is also an SRO. The purpose as established in NUREG-0585 was to assure the availability of at least one qualified SRO in the control room without affecting the freedom of the shift supervisor to move about the site as needed. This requirement, which in effect calls for an additional SRO per shift per site is to be met by July 1, 1982, at all operating stations.

Preliminary assessment of the status of 49 currently operating stations indicates that 12 licensees have already met the requirement; compliance by 16 other is probable; 7 have formally requested extensions beyond July 1, 1982; 6 have stated that meeting the deadline is contingent on such factors as candidates passing upcoming license examinations; 4 have proposed combining the position of shift technical advisor with that of the additional SRO; and 4 cannot or are not planning to meet the deadline.

Licensees cite a number of reasons for not being able to meet the deadline. Foremost among these is the time required to train individuals to become SROs. It can take four or more years to select and hire qualified individuals, train them, and provide needed experience prior to their qualifying as SROs. This total time could well be required even though the actual training of an experienced reactor operator (RO) to become an SRO occupies only about one year, since the RO cannot be relieved of his duties until a replacement has been trained and qualified for his position. This problem is particularly acute for those licensees who, prior to the TMI-2 accident, were operating with minimum crews and without a sufficient pipeline of operator trainees.

Another problem cited by licensees is a turnover rate that is higher than anticipated. Uncertainty regarding proposed regulatory requirements for operator qualifications has resulted in many licensed operators opting for different jobs. Losses of licensed operators are also being experienced as utilities bid for personnel who have had operating experience to help staff plants that are soon to be licensed. Furthermore, increased training burdens and more stringent operator license examinations have resulted in greater attrition in operator training programs.

INTEGPATED PLANT SAFETY ASSESSMENT FOR PALISADES

Division of Licensing

By Theodore Michaels, Systematic Evaluation Program Branch

The first of ten safety assessments being performed under the Systematic Evaluation Program (SEP) Phase II was completed and issued as NUREG-0820 (DRAFT) "Integrated Plant Safety Assessment, Palisades Plant."

The SEP was initiated by the NRC to review the designs of older operating nuclear reactor plants to reconfirm and document their safety. The review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

Under Phase I of the SEP, 137 different areas of review (topics) were derived and under Phase II of the SEP these topics are being reviewed to compare the as-built plant design with current review criteria for the following plants in addition to Palisades: Big Rock Point, Dresden 2, Ginna, Haddam Neck, LaCrosse, Millstone 1, Oyster Creek, San Onofre, and Yankee.

During the review, 47 of the topics were deleted from consideration by the SEP because a review was being made under other programs (Unresolved Safety Issue [USI] or Three Mile Island [TMI] Action Plan Tasks), or the topic was not applicable to the plant; that is, the topic was applicable to boiling-water reactors rather than to pressurized-water reactors. The status of the USI and TMI tasks will be addressed in a supplement to NUREG-0820 and will be available as one of the bases for considering the conversion of the provisional operating license for Palisades to a full-term operating license.

Of the original 137 topics, 90 were therefore reviewed for Palisades to determine whether the corresponding plant design was consistent with current licensing criteria contained in regulations, guides, and the Standard Review Plan or the equivalent of such criteria. Of these topics, 57 met current criteria or were acceptable on another defined basis. Additionally, two topics were found acceptable as a result of modifications made by the licensee during topic review. Parts of three other topics were also found acceptable as a result of modifications made by the licensee during topic review; other parts of these topics did not meet current criteria and were considered in the integrated assessment.

The review of the 31 remaining topics found that certain aspects of plant design differed from current criteria. These topics were considered in the integrated assessment of the plant, which consisted of evaluating the safety significance and other factors of the identified differences from current design to arrive at decisions on whether backfitting was necessary from an overall plant safety viewpoint. To arrive at these decisions, engineering judgment was used as well as the results of a limited probabilistic risk assessment study. Backfit requirements fell into one or more of the following categories:

(1) Equipment modification or addition,

(2) Procedure development or changes, and

(3) Refined engineering analysis or continuation of ongoing evaluation.

In general, it can be concluded that plant design conformed well or was equivalent to new plant requirements for areas reviewed.

A review of the operating history up to 1979 is also included in the report as well as recent enforcement history. The operating history continues to involve regulatory noncompliance related to failure to adhere to procedure or human error events through early 1981. Early in 1981, the company developed and implemented a program for the improved regulatory performance at the Palisades Plant. The program involved administrative reorganization, increasing the staff, procedural review and revision, increased training efforts and other functions. As result of this program, the regulatory performance since early 1981 has shown improvement.

RELIABILITY ANALYSIS OF EMERGENCY AC POWER SYSTEM

Division of Safety Technology

Task Manager for Generic Issues Branch is Patrick Baranowsky of the Office of Nuclear Regulatory Research

Unresolved Safety Issue A-44 is concerned with the likelihood and the potential accident risks of a station blackout, which means the complete loss of AC power. An analysis has been made at the Oak Ridge National Laboratory of the reliability of the onsite or emergency AC power system. The reliability of the offsite power system is being analyzed separately.

The emergency AC power system may consist of one, two, three, or four diesel generators dedicated to a single nuclear power unit, or diesel generators may be shared among units at a given site. Responses to questionnaires on operating experience were received for 45 units having 194 reactor years and 450 diesel-generator years of operation. Of 13,885 total demands (including testing) on diesel generators, there were 271 failures. Eighteen plants were selected as representative of the spectrum of system designs and operating experience, and fault-tree models were developed and analyzed for each of these plants.

A preliminary and generic evaluation indicates that a two-reactor plant requiring two out of three diesels to function to properly cool both units has a calculated frequency of station blackout that is five times as great as a similar plant requiring two out of four diesels to function. A plant with diesels cooled by service water may have a frequency of station blackout of up to three times the frequency for a plant with air-cooled diesels.

Control power from a DC bus is necessary for a diesel generator to provide emergency AC power to the plant. In some cases, the diesel has a dedicated battery but requires both the dedicated battery and the plant (Class 1E) battery to start and load the diesel. In those cases, the dedicated battery may provide an additional failure mode and the reliability of the diesel could be lower than it would be if there were no battery dedicated to the diesel.

The average number of failures of a diesel generator is three per hundred demands based on actual experience with loss of offsite power and safety injection events. Other statistical data and a breakdown of failures by subsystem are given on the next page. The unavailability of diesel generators for scheduled maintenance is a significant contributor to unreliability at some plants. Operating and maintenance procedures that are not written with sufficient clarity, direction, and/or verification checks are also contributors and increase the likelihood of common-cause failure.

There are improvements that can be made in the diesel generator subsystems, but there are no one or two modifications that can be made at all plants to decrease the industry-average failure rate significantly. The number of equipment and design modifications that have been made to correct the cause of a failure at a plant and to improve the operator's ability to monitor and control the diesel ranges from one in five years to about four a year for five years. There have been several design changes by some vendors to improve diesels for nuclear plant service.

DIESEL GENERATOR FAILURES

Statistical Data

Failures per demand (actual*)	0.03
Failures per demand (all**)	0.02
Failure to run ***	0.0024 per hour
Mean time to repair	24 hours
Unavailability due to testing and maintenance	0.006

Failure by Subsystem

Subsystem	Percent of Reported Failures
Logic and control	14.8
Governor	12.4
Breaker and sequencer	10.3
Fuel	9.3
Cooling	11.6
Start	9.5
All others	32.1

Cause of Failure

Human error	~ 20 percent
Design and hardware	~ 80 percent

* Actual experience with loss of offsite power and safety injection events.

** Includes testing.

*** Failure rate per hour assuming diesel is started and loaded.

REMOTE AREA MONITORING SYSTEMS AT POWER REACTORS

Division of Systems Integration

Contact is Seymour Block, Radiological Assessment Branch

Remote area monitors (RAMs) are used to measure ambient radiation levels on a continuous basis. A RAM system consists of one or more radiation detectors and associated electronics hard-wired to a readout assembly in another location or room, perhaps several hundred feet away. RAM systems may be provided also with local readout at or near the detector, internal calibration capability, logarithmic or autoranging readout to cover a wide range of readings, and local and remote alarm capability to indicate when a preset radiation level has been exceeded. Important applications include monitoring of areas with actual or potential high ambient radiation levels to provide warning, without exposure of personnel, when abnormal levels or malfunction of equipment occur.

At the request of the NRC, the Pacific Northwest Laboratory sent out a questionnaire to 68 licensed nuclear power plants regarding installed RAM systems. Returns were received from 55 of the plants, 17 boiling water reactors (BWRs) and 38 pressurized water reactors (PWRs). A total of 91 separate systems were reported, which were manufactured by 10 different companies, with three companies supplying more than 75% of the RAMs.

The type, location, and number of detectors in all the reported BWRs and PWRs are given in the table on the next page. The average number of detectors is 35 per BWR and 20 per PWR. For detectors in containment, five of the BWRs did not report any and the remainder had an average of 15, while all PWRs reported some and had an average of three. A problem of nomenclature and variation in plant structures may have contributed to the diversity of the responses.

The upper limit of the capability of RAMs in containment ranged from 10 or less to 1,000 roentgens per hour for BWRs and from one or less to 10,000,000 roentgens per hour in PWRs. Most of the reporting plants had local and control-room readouts. None of the BWRs and only 14% of the PWRs reported a readout in the emergency operations center. Audible and visual high-level-radiation alarms were provided in all but one of the control rooms. All reporting plants had local alarms (near the detector).

Forty-two of the plants reported using a radioactive source to make routine operational checks. Calibration was performed at all the plants and was done in place, except for three plants where the calibration was done in the health physics laboratory.

Most of those responding were generally pleased with the performance of the RAMs in their plants. Some offered suggestions regarding additional features they deemed desirable. Several noted that upgrading of various RAM systems was in progress or planned.

Further information may be found in NUREG/CR-2413, "Survey of Remote Area Monitoring Systems at U.S. Light-Water-Cooled Power Reactors," published in April 1982.

Type and Number of Detectors by Location in BWRs

Type of Detector	Containment	Auxiliary Building	Fuel Handling	Turbine Generator	Control Room	Other	Total No. of Type	% of Total
	Number at Location Specified							
G-M	181	102	103	46	115	28	575	97%
Ion chamber	0	0	0	1	0	0	1	0.2%
Scintillation	2	6	2	0	0	7	17	3%
Other	0	0	0	0	0	0	0	0
Total	183	108	105	47	115	35	593	100%

Type and Number of Detectors by Location in PWRs

Type of Detector	Containment	Auxiliary Building	Fuel Handling	Turbine Generator	Control Room	Other	Total No. of Type	% of Total
	Number at Location Specified							
G-M	65	250	40	29	23	23	430	57%
Ion chamber	40	97	14	10	10	10	181	24%
Scintillation	10	84	23	6	8	12	143	19%
Other	0	0	0	0	0	0	1	0%
Total	116	431	77	45	41	45	755	100%

DETECTION OF LOOSE PARTS IN THE STEAM GENERATOR

Division of Systems Integration

By Yi-hsiung Hsii, Core Performance Branch

Section 4.4 of the Standard Review Plan (NUREG-0800) requires that each plant incorporate a loose-parts monitoring system (LPMS). The acceptance criteria states that the design description and proposed procedures for use of the loose parts monitoring system should be consistent with the requirements of Regulatory Guide 1.133. The primary purpose of a LPMS is for early detection of loose metallic parts in the primary system of light-water cooled reactors.

To detect loose parts, a LPMS relies on sensors, called accelerometers, to monitor acoustic disturbances generated by metal impact. An effective LPMS must have an adequate number of properly deployed sensors strategically located on the exterior surface of the reactor coolant pressure boundary. The number and locations of the sensors depend on the functional requirements placed on the LPMS. Regulatory Guide 1.133 recommends that a minimum of two sensors be located at each region prone to the natural collection of drifting objects, such as the reactor vessel upper and lower plena and the steam generator inlet plenum. These sensors will detect the presence of loose parts and will be indicative of their general locations. More precise impact location and loose part characterization are much more costly and difficult goals to achieve and are not required.

To avoid or minimize false alarms caused by background mechanical and hydraulic noises, the systems sensitivity is limited. Regulatory Guide 1.133 recommends that the on-line sensitivity of the automatic detection system be capable of detecting a loose metallic part that weighs from 0.25 to 30 pounds and impacts with a kinetic energy of 0.5 foot-pounds on the inside surface of the reactor coolant pressure boundary within 3 feet of a sensor. This sensitivity criterion is readily achievable under the acoustically quiet condition; but the system needs proper calibration to account for background noise during normal reactor operation. Some LPMSs automatically adjust alert level as a function of background noise and therefore alleviate the number of false alarms.

Although its primary purpose is for the primary-side loose parts detection, the sensors placed on the inlet plenum of a steam generator can also detect loose part impacts in the secondary side. However, no data are available to demonstrate quantitatively the effectiveness of LPMS on the detection of loose parts in the secondary side. A loose part in the secondary side may lodge on the tube sheet or between tube support plates at various heights. The effectiveness of the sensors to detect the impact noise depends on the acoustic coupling of the sensors to the sound wave transmission paths. Increasing the number of sensors with spatical coverage on the steam generator may provide additional sentitivity for detection of loose parts far away from the inlet plenum. These sensors should be mounted in the locations acoustically coupled to detect metallic impacts on the steam generator tubes and tube sheet and support plates. The sensor mounting method plays a very important role in the sensor effectiveness. Makeshift mounting schemes such as by magnets or adhesives should be avoided.

In summary, a LPMS conforming to Regulatory Guide 1.133 guidelines should be an effective indicator of loose parts in the steam generator secondary. The effectiveness may be increased by inclusion of additional sensors at strategic locations.