RAS 1022





October 5, 1999



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

| | | |
|---|---|---|
| In the Matter of | ) | |
| | ) | Docket No. 50-400-LA |
| CAROLINA POWER & LIGHT | ) | |
| COMPANY | ) | ASLBP No. 99-762-02-LA |
| | ) | |
| (Shearon Harris Nuclear Power Plant) | ) | |
| | ) | |

NRC STAFF'S RESPONSE TO ORANGE COUNTY'S FIRST SET OF DISCOVERY REQUESTS TO NRC STAFF

The Nuclear Regulatory Commission staff (Staff) hereby responds to Orange County's First Set of Discovery Requests to NRC Staff, filed September 20, 1999.

The Staff notes that 10 C.F.R. §§ 2.744 and 2.790, which govern the production of NRC records and documents, contemplate that most NRC documents will be available for inspection and copying in the public document room, and, if they have been withheld from the public document room pursuant to § 2.790, a request to the Executive Director for Operations for the production of such a document is required by § 2.744, which must state, among other things, why the requested record or document is relevant to the proceeding.

Notwithstanding these regulations, and in accordance with a September 23, 1999 agreement of counsel, without waiving any objections or privileges, and except as specified below, the Staff is now voluntarily providing responses to Orange County's request for production of documents. In doing so, the Staff is not waiving its right to require full

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SECY-02

compliance with the Commission's regulations regarding any future discovery requests made by Orange County in this matter.

## I. GENERAL OBJECTIONS

1. The Staff objects to Orange County's discovery requests to the extent that they call for disclosure of litigation strategy and other material protected under 10 C.F.R. § 2.740 or other protection provided by law, attorney work product, privileged attorney-client materials, and other privileged materials such as draft agency documents protected by executive privilege.

2. The Staff objects to Orange County's discovery requests to the extent that they request information or documents relating to licensees and/or entities other than Carolina Power & Light's Shearon Harris Nuclear Plant. Such discovery requests call for information which is irrelevant, immaterial, and not calculated to lead to the discovery of admissible evidence, and are over-broad and unduly burdensome.

3. The Staff objects to Orange County's discovery requests to the extent that they require identification of the home addresses and telephone numbers of Staff employees or contractors, which are protected from disclosure by the Privacy Act, 5 U.S.C § 552a(b) and 10 C.F.R. § 2.790(a)(6). The disclosure of such information is irrelevant and unnecessary.

4. The Staff objects to Orange County's discovery requests to the extent that they seek discovery which is beyond the scope of the two contentions admitted by the Board in this proceeding. Orange County is only permitted to obtain discovery of matters that pertain to the subject matter within the scope of this proceeding.

## II. GENERAL DISCOVERY REQUESTS

### A. GENERAL INTERROGATORIES

Pursuant to agreement between Orange County and the Staff, these general interrogatories apply to both Orange County admitted contentions; are in addition to the fifteen interrogatories per contention allowed by the Board's July 29, 1999, Memorandum and Order; and are continuing in accordance with 10 CFR § 2.740(e) through the end of the discovery period, October 31, 1999, as established in the Board's July 29, 1999 Memorandum and Order.

GENERAL INTERROGATORY NO. 1. State the name, business address, and job title of each person who supplied information for responding to these interrogatories, requests for admission, and requests for the production of documents. Specifically note for which interrogatories and requests for admissions each such person supplied information. For requests for production, note for which contention each such person supplied information.

STAFF'S RESPONSE: The following persons supplied information for responding to Orange County's First Discovery requests:

Richard Laufer
Project Manager, Shearon Harris Nuclear Power Plant
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Document Requests for both contentions

Lawrence Kopp
Senior Reactor Engineer
Reactor Systems Branch, Division of Systems Safety Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Document Request for contention 2

Kenneth C. Heck
Quality Operations Engineer
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Document request for contention 3

Donald G. Naujock
Technical Reviewer
Materials and Chemical Engineering Branch, Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Document Request for contention 3

James A. Davis
Technical Reviewer,
Materials and Chemical Engineering Branch, Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Document request for contention 3

GENERAL INTERROGATORY NO. 2. For each admitted Orange County contention, give the name, address, profession, employer, area of professional expertise, and educational and scientific experience of each person whom the NRC Staff expects to provide sworn affidavits and declarations in the written filing for the Subpart K proceeding described in the Board's July 29, 1999, Memorandum and Order and the general subject matter on which each person is expected to provide sworn affidavits and declarations for the written filing. For purposes of answering this interrogatory, the educational and scientific experience of expected affiants and declarants may be provided by a resume of the person attached to the response.

The Staff has not yet made a final determination regarding who will provide sworn affidavits, but provides the following as persons who are likely to provide a sworn affidavit or declaration in the written filing for the subpart K proceeding:

> Richard Laufer
> Project Manager, Shearon Harris Nuclear Power Plant
> General subject matter: The overall project
>
> Lawrence Kopp
> Senior Reactor Engineer
> General subject matter: Contention 2
>
> Kenneth C. Heck
> Quality Operations Engineer
> General subject matter: Contention 3
>
> Donald G. Naujock
> Technical Reviewer
> General subject matter: Contention 3
>
> James A. Davis
> Technical Reviewer
> General subject matter: Contention 3
>
> Anthony Ulses
> Nuclear Engineer
> Reactor Systems Branch, Division of Systems Safety and Analysis
> Office of Nuclear Reactor Regulation
> U.S. Nuclear Regulatory Commission
> Washington, D.C. 20555
> General subject matter: Contention 2

A copy of the resume of each person named in this answer is annexed hereto as attachment 1. The Staff reserves the right to amend this answer as discovery continues.

> <u>GENERAL INTERROGATORY NO. 3.</u> For each admitted Orange County contention, identify each expert on whom the NRC Staff intends to rely on in its written filing

for the Subpart K proceeding described in the Board's July 29, 1999 Memorandum and Order, the general subject matter on which each expert is expected to provide sworn affidavits and declarations for the written filing, the qualifications of each expert whom the NRC Staff expects to provide sworn affidavits and declarations for the written filing, a list of all publications authored by the expert within the preceding ten years, and a listing of any other cases in which the expert has testified as an expert at a trial, hearing or by deposition within the preceding four years.

Lawrence Kopp
List of publications is contained in attached resume
General subject matter: Contention 2; criticality

Kenneth C. Heck
General subject matter: Contention 3

Donald G. Naujock
Publication listed in attached resume.
General subject matter: Contention 3

James A. Davis
List of publications attached.
General subject matter: Contention 3

The Staff reserves the right to amend this answer as discovery continues.

B. GENERAL DOCUMENT REQUESTS

The County requests the Staff to produce the following documents directly or indirectly within its possession, custody or control.

> REQUEST NO 1. All documents in your possession, custody or control that are identified, referred to or used in any way in responding to all of the above general interrogatories and the following interrogatories and requests for admissions relating to specific contentions.

STAFF'S RESPONSE: No documents, other than the attached resumes, were used in answering the general interrogatories.

> REQUEST NO. 2. All documents in your possession, custody or control relevant to each Orange County admitted contention, and to the extent possible, segregated by contention and separated from already produced documents.

STAFF RESPONSE: All available, non-objectionable, responsive documents that are relevant to the two contentions, as admitted into the proceeding by the Board, which are not in the Public Document Room (PDR) or have not been previously produced will be provided in response to this request either with this document or within 30 days of the date of Orange County's first Discovery Request to the NRC Staff. Documents which are generally available to the public will not be produced. A list of responsive documents served together with this response is annexed hereto as attachment 2. The Staff reserves the right to amend this answer as discovery continues.

> REQUEST NO. 3. All documents (including experts' opinions, workpapers, affidavits, and other materials used to render such opinion) supporting or otherwise relating to

> testimony or evidence that you intend to use in your Subpart K presentation and/or the hearing on each Orange County admitted contention.

STAFF RESPONSE: The Staff objects to this document request as being overly broad, unduly burdensome, seeking pre-decisional, trial preparation or privileged material or material exempted from disclosure by 10 C.F.R. §§ 2.744 and 2.790. Without waiving these objections, any available, non-objectionable documents that are relevant to the two contentions, as admitted into the proceeding by the Board, which are not in the Public Document Room (PDR) or have not been previously produced will be provided in response to this request within 30 days of the date of Orange County's first Discovery Request to the NRC Staff. The Staff reserves the right to amend this answer as discovery continues.

## IV. SPECIFIC DOCUMENT REQUESTS

Subject to the limitations specified on page 1 of this document, the Staff will respond to Orange County's specific document requests within 30 days of receipt of Orange County's First Discovery Requests.

Respectfully submitted,

Susan L. Uttal
Counsel for NRC staff

Dated at Rockville, Maryland
this 5th day of October 1999

# RESUME

James A. Davis
Security Clearance: Q (Top Secret)

Education:
High School
Worthington High School, Worthington, Ohio
College
B.Met.E., The Ohio State University, 1965, Metallurgical Engineering
M.S., The Ohio State University, 1965, Metallurgical Engineering
Ph.D, The Ohio State University, 1968, Metallurgical Engineering
39/45 Credits towards an MBA, Canisius College, 1972-1976.

Experience:
11/11/1990 to Present: U.S. Nuclear Regulatory Commission, GG-14-10
Supervisor: E. Sullivan
Mailing Address: U.S. NRC
Mail Stop O7-D4
Washington, D.C. 20555

I am a technical reviewer in the Chemical Engineering and Metallurgy Section of the Division of Engineering, the Office of Nuclear Reactor Regulation. Areas of responsibility include coatings for nuclear power plants, license renewal for Calvert Cliffs and Oconee, all threaded fastener issues (such as stress corrosion cracking, boric acid corrosion, and fatigue), chemical decontamination, Boiling Water Reactor internals cracking, pump and valve internals cracking, pipe integrity issues, and corrosion behavior for dry cask storage, and interaction of coatings with spent fuel water, Currently, I am coordinating the responses to a generic letter on containment coatings for nuclear power plants. I am the NRC representative to ASTM D-33 on coatings for power generation facilities. I am also a member of the Board of Directors for the National Board of Registration for Nuclear Safety Related Coating Engineers & Specialists. I am also a member of ASME B1 on threaded fasteners. I was a member of an Augmented Inspection Team at Palisades on fuel handling problems and Point Beach on the hydrogen burn as a result of interactions between borated water and the inorganic Zinc coating during dry cask loading operations. I was Contract Technical Monitor and Project Officer for numerous contracts at Brookhaven National Labs. I was a technical reviewer for the design of the Navy Seawolf Submarine and on the DOE project to produce tritium in a commercial reactor (Watts Bar). I was acting section chief on numerous occasions, and for several months at a time. I have made numerous presentations to senior NRC management including the Chairman, the Executive Director for Operations, the Committee to Resolve Generic Issues, and the Advisory Committee on Reactor Safety and Safeguards. I testified before Representative Dingle's staff on the safety of fasteners in nuclear power plants as a result of concerns raised by a private citizen. I convinced his staff that there is no safety issue because of the redundant design of mechanical joints, the fact that the joints will leak before they break, and that the joints are inspected every refueling outage.

8/1981 to 4/1990: Polyken Technologies/Kendall Co., Senior Research Associate
Supervisor: Jordan Kellner
Address: Polyken Technologies/Kendall Co.
17 Hartwell Avenue
Lexington, MA 02173

I was responsible for domestic and international technical marketing for Polyken Pipeline Coatings. As part of my job, I made technical presentations on Polyken Pipeline Coatings in North America, USSR, Egypt, India, Iraq, Japan, Australia, Bolivia, France, England, Germany, Czechoslovakia, Italy, Switzerland, Algeria, Singapore, and Jakarta to ministries and high level government officials. I coordinated joint research between Polyken and the VNIIGAS and VNIIST Technical Institutes in Moscow on the development of high temperature pipeline coatings. I contracted with independent laboratories to certify Polyken products for international customers. I was the Polyken representative to National Association of Corrosion Engineers (NACE) committees on Underground Pipeline Coatings, Arctic Corrosion, and Cathodic Protection. I was appointed by the President of National Association of Corrosion Engineers to the International Relations Committee. Also, I was the company representative to American Society of Testing and Materials and the American Water Works Association technical committees. I was responsible for analyzing competitive coatings using fast Fourier transform infrared spectroscopy, gas chromatography, mass spectroscopy, scanning electron microscopy, differential scanning calorimetry, thermal gravimetric analysis, moisture vapor transmission apparatus, and mechanical test equipment. I conducted slow strain rate stress corrosion cracking tests on line pipe steel to develop inhibitors that could be added to coating primers to control stress corrosion cracking. I conducted slow strain rate tests on various blends of polypropoline to develop blends that are resistant to environmental cracking. I was acting section chief for extended periods of time.

11/1979 to 8/1981: Arthur D. Little, Senior Consultant
Supervisor: William Lee
Address: Arthur D. Little
15 Acorn Park
Cambridge, MA 02138

I was a consultant to DOE on the metallurgical and mechanical condition of defense nuclear waste tanks that were damaged during stress relieving and that were damaged by sulfate reducing bacteria. This included a review of the structural integrity of the waste tanks and a review of DuPont's program to control stress corrosion cracking of the tanks. I was a consultant to DOE on the Savannah River Defense Waste Form program. Twelve contractors were developing defense waste forms including borosilicate glass, high silica glass, synrock, coated borosilicate glass, HIP rock, and stoichiometric concrete. A pilot plant borosilicate glass facility was constructed at Savannah River. My job was to visit with each contractor twice a year and review their programs. I then reviewed progress reports and prepared an assessment of each contractor's work and recommended that individual programs be expanded, contracted, or canceled. I was also a member of a team that developed models for long term storage of commercial nuclear waste sponsored by Battelle. I consulted to numerous commercial customers on corrosion, fracture mechanics, coating, metallurgical, and plating issues.

11/1978-11/1979: Allied Tube and Conduit Corp., Director of Research
Supervisor: L. Volmuth
Address: Allied Tube and Conduit
16100 South Lathrop
Harvey, IL 60426

I was responsible for research and development in the areas of metallurgical tube forming, low frequency electric resistance welding, chemical cleaning of steel, high speed in-line galvanizing, surface treating, and coating of electrical conduit, fence posts, and specialty tubing. I was also responsible for the Quality Control Department and the Process Control Laboratory.

6/1976-11/1978: Allegheny Ludlum Steel Corp., Research Specialist
Supervisor: George Aggen
Address: Allegheny Ludlum Steel Corp.
Research Center
Brackenridge, PA 15014

I determined the level of columbium and vanadium stabilizers required to avoid sensitization of ferritic stainless steels. I developed intergranular sensitization tests for low chromium ferritic stainless steels. I used electrochemical and electron-optical including scanning electron microscopy, auger, and transmission electron microscopy techniques for failure analyses. I used sensitive analytical techniques to determine carbon, nitrogen, columbium, and titanium contents of stainless steel. I provided customer service by recommending specific grades of stainless steel for corrosive applications.

11/1970-6/1976: Bell Aerospace: Senior Research Scientist
Supervisor: A. Watts
Address: Bell Aerospace
Wheatfield, NY

I was program Manager on numerous Navy sponsored programs involving the corrosion of aluminum alloys, stainless steels, and titanium alloys in high velocity sea water for the Navy's high performance ships program. I examined the influence of mean stress intensity and stress intensity amplitude on the corrosion fatigue behavior of aluminum alloys and titanium alloys in sea water using fracture mechanics specimens. I determined the comparability of various materials with cooling fluids for the hypersonic airplane. I examined the comparability of rocket fuels and oxidizers with fuel handling equipment. I managed the fracture mechanics group. I developed microelectrodes for measuring potential and pH inside of growing stress corrosion cracks of aluminum alloys and alloy steels.

1/1968-11/1970: U.S. Steel Corporation, Senior Research Engineer
Supervisor: Brian Wilde
Address: U.S. Steel Corp
Jamison Lane
Monroeville, PA

I developed steel with improved corrosion resistance using linear polarization, anodic polarization, transmission electron microscopy, and scanning electron microscopy. I conducted fundamental studies on the mechanism of pitting, stress corrosion cracking, hydrogen embrittlement, and intergranular corrosion using electrochemical techniques, static and dynamic straining techniques, hydrogen permeation cells, and optical and scanning microscopy techniques. I was the group leader of the electrochemistry group.

NRC Awards:
Performance Award - December 21, 1994
Certificate of Appreciation - February 1, 1995 - Seawolf Class Submarine design review
Group Award - December 10, 1996 - AIT Team at Point Beach
Performance Award - January 10, 1997

NRC Training:
Power Plant Engineering (2 week course)
Boiling Water Reactors-General Electric Design (1 week course)
Pressurized Water Reactors-Westinghouse Design (1 week course)
Non Destructive Testing (2 week course)
Sexual Harassment Awareness (2 day course)
AIDS Awareness (1 day course)
Allegation Training (4 hours annually)
Procurement Training (1 day)
Ethics Training (4 hours annually)
Security Training for the Seawolf Design Review (4 hours)
Security Training for the DOE tritium Project (4 hours)
NUDOCS (NRC document retrieval system) (1 day)
dBase III (2 days)
Autos LAN Training (4 hours)
Introduction to Netscape
Introduction to Microsoft NT
Introduction to WordPerfect 8.1

Other Relevant Training
Public Speaking-Kendall-3 Days, Allegheny Ludlum-5 days
Effective Writing-U.S. Steel(3 days)

**PUBLICATIONS-PRESENTATIONS**

1 **J. A. Davis and J. D. Kellner, "Electrochemical Principals Applied to Operating Pipelines," Presented at Corr/89, Underground Corrosion Symposium, New Orleans, Louisiana, April 17-21, 1989**

2. **F. J. Witt, J. A. Davis and R. A. Hermann, "Second EPRI Balance-of-Plant Heat Exchanger NDE Workshop, The U. S. Nuclear Regulatory Commission Perspective," Second EPRI Balance-of-Plant Heat Exchange NDE Workshop, Key West, Florida, May 26-29, 1992**

3. **J. A. Davis, "Full Reactor Coolant System Chemical Decontamination, NRC Approval of the Westinghouse Owners Group Topical Report" Fifth EPRI**

Workshop on Chemical Decontamination, Charlotte, North Carolina, June 8-9, 1993

4. James A. Davis and Richard E. Johnson, "The Regulatory Approach to Fastener Integrity in the Nuclear Industry," Symposium on Structural Integrity of Fasteners, Sept. 8-19, 1993, to be published as a Special Technical Publication.

5. James A. Davis, "Nuclear Power Plant Service Water Systems, NRC Staff Perspective," International Joint Power Generation Conference, ASME Power Division, Heat Exchange Committee, Kansas City, Mo., October 17-21, 1993

6. J. A. Davis, G. P. Hornseth, and R. A. Hermann, "Third EPRI Balance of Plant Heat Exchanges NDE Workshop, A Regulatory Perspective," Third EPRI Balance of Plant Heat Exchange Workshop, Myrtle Beach, SC., June 6-8, 1994

7. R. A. Hermann, M. Banic, and J. A. Davis, "Primary Water Stress Corrosion Cracking of Alloy 600," Specialists Meeting on Cracking in LWR RPV Head Penetrations, International Atomic Energy Agency, Philadelphia, PA, May 2-4, 1995

8. Robert A. Hermann, James A. Davis, and Merrilee J. Banic, "Age Related Degradation in Operating Nuclear Plants," presented at ASME International Vessel and Piping Conference, Honolulu, Hawaii, July, 1995

9. James A. Davis, "A Regulatory Prespective on Service Water Problems," Invited Lead Speaker, Fourth EPRI Balance-of-Plant Heat Exchanger NDE Symposium, June 10-12, 1996, Jackson Hole, Wyoming.

10. James A. Davis, "Nuclear Power Plant Fastener Thread Gaging - NRC Staff Perspective," presented at Eight International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, August 10-14, 1997, Amelia Island, Florida.

11. James A. Davis, "A Regulatory Prespective on Service Water Problems," Invited Lead Speaker, Fifth EPRI Balance-of-Plant Heat Exchanger NDE Symposium, June 15-17, 1998, Lake Tahoe, Nevada.

# Anthony P. Ulses

301 415 1194

*apu@nrc.gov*


## Education

Master's of Science (MS) in Nuclear Engineering
University of Maryland at College Park, May 1999.

Bachelor's of Science and Engineering (BSE) in Nuclear Engineering, *Cum laude*
University of Michigan at Ann Arbor, August 1992.

## Experience

United States Nuclear Regulatory Commission — 5/93-present
Nuclear Engineer, Reactor Systems Branch, Division of Systems Safety and Analysis

United States Nuclear Regulatory Commission — 9/92-5/93
Graduate Fellow, Advanced Reactors Project Directorate

*Computer Code Development*
- Maintained and upgraded legacy physics codes on UNIX workstations
- Developed VIKTORIA code for fuel channel analysis
- Coupled TRAC and NESTLE codes
- Assisted in NEWT development

*Computer Code Analysis*
- SCALE — Fuel lattice criticality studies, depletion and collapsed cross section preparation
- MCBEND — Reactor pressure vessel fluence studies
- NEWT — Fuel power distributions, depletion and collapsed cross sections
- DOORS 3.2 — 3D Transport Calculations
- TRAC/NESTLE — 3D BWR ATWS Studies
- DRAGON 3.2 — Fuel power distributions, depletion and collapsed cross sections

*Computer Codes*
- DANTSYS 3.1, MONK

*Computer Languages and Operating Systems*
- UNIX, Fortran 90/95, Fortran 77, C, Windows, DOS, PVM

*Regulatory Experience:*
- License amendment evaluations
- Fuel manufacturer inspections
- Performing Audit Calculations of Licensee Analyses

*General Experience:*
- Managing High Performance Computer Networks
- Digital UNIX System's Administration

Laurence I. Kopp
Senior Reactor Engineer

## Education

Ph.D., Nuclear Engineering, University of Maryland, 1968
M.S., Physics, Stevens Institute of Technology, 1959
B.S., Physics, Fairleigh Dickinson College, 1956.

## Employment

U.S. Nuclear Regulatory Commission, Senior Reactor Engineer, 1965 - present
Performs safety evaluations of reactor license applications, technical specifications, core reloads, spent fuel storage facilities, and topical reports. Developed regulatory guides, information notices, generic letters, rulemaking related to reactor physics, safety analysis, and fuel storage. Assisted in development of improved technical specifications in areas of reactivity control, power distribution limits, and fuel storage.

Westinghouse Astronuclear Laboratory, Senior Scientist, 1963-1965
Evaluated nuclear analytical methods to be used in the design of NERVA rocket reactors. Analyzed experiments performed in the Los Alamos zero power reactor.

Martin-Marietta Nuclear Division, Senior Engineer, 1959-1963
Performed core physics calculations on fluidized bed and PM-1 reactors. Performed parametric studies of reactors applicable to nuclear rocket applications. Programmed several FORTRAN computer codes.

Federal Electric Corporation, Senior Programmer, 1957-1959

Curtiss-Wright Research Division, Programmer/physicist, 1956-1957

## Professional Societies

American Nuclear Society
ANS-10 Mathematics and Computations Standards Committee
ANSI N-17 Standards Committee on Research Reactors, Reactor Physics & Radiation Shielding

## Publications

"The NRC Activities Concerning Boraflex Use in Spent-Fuel Storage Racks," invited paper, American Nuclear Society Annual Meeting, June 1996.

"Potential Loss of Required Shutdown Margin During Refueling Operations," invited paper, American Nuclear Society Annual Meeting, June 1990.

"Recommended Programming Practices to Facilitate the Portability of Scientific Computer Programs," ANS Proceedings of the Topical Meeting on Computational Methods in Nuclear Engineering, April 1979.

"The Neutron Resonance Integral of Natural Dysprosium " Ph.D. thesis, University of Maryland, 1968.

"Pool Reactor Experiments with Control Rods," Transactions of the American Nuclear Society, Vol. 10, Pg. 16, 1967 (co-author).

"Procedures for Obtaining Few-Group Constants for Systems Having Rapid Spectral Variation With Position," Transactions of the American Nuclear Society, Vol. 8, pg. 303, 1965 (co-author).

"Improved Nuclear Design Method for NERVA Calculations - NSDM II, WANL-TME-1091, Westinghouse Astronuclear Laboratory, 1965 (co-author).

"Analysis of Experiments Performed in Los Alamos ZEPO Reactor," WANL-TME-273, Westinghouse Astronuclear Laboratory, 1963.

**KENNETH C. HECK**
**735 University Avenue**
**Sewanee, Tennessee 37383**

**Tel: (301) 415-2682**
**email: kch1@nrc.gov**

## SUMMARY OF SKILLS

Technical, supervisory, and management experience in the electric power industry and with assignments in engineering, project management, project engineering, plant start-up, plant and program evaluation, quality assurance, and licensing. Proficiencies include design engineering, control systems, electronics, accounting, and computer applications.

## EXPERIENCE

Nuclear Regulatory Commission (May 1997 - Present)

**Quality Operations Engineer (Headquarters), Inspection Program Management**
- Review, evaluate, audit quality assurance programs and other administrative control aspects for nuclear power plants.
- Perform program development functions related to all aspects of the agency's quality assurance programs.
- Conduct inspections of vendors who provide products and services to the nuclear industry.

Tennessee Valley Authority

**Lead Auditor, Quality Services** (June 1995-October 1996)
- Provided staff augmentation services in the areas of quality assurance and licensing.
- Developed audit/consultation services for implementing international (ISO-9000) quality standards.

**Principal Evaluator, Nuclear Assurance & Licensing** (October 1988-June 1995)
- Conducted independent audits/evaluations of nuclear power programs, processes, and plant events.
- Served as Technical Secretary for the Nuclear Safety Review Board (senior safety oversight body) from shutdown of TVA's nuclear program through recovery of the Sequoyah and Browns Ferry nuclear plants.
- Conducted independent verifications of the effectiveness of completed corrected actions through successful startup of the Watts Bar nuclear plant.

**Senior Evaluator, Nuclear Managers Review Group** (March 1987 to October 1988)
- Developed and implemented a review program to assess activities associated with the design, construction and operation of TVA nuclear plants. Findings were reported directly to the Manager, Nuclear Power with recommendations for improvements.

Independent Contractor (December 1985-March 1987)

**Design Engineer/System Engineer, Engineering Department**
- Modified the integrated control system and non-nuclear instrumentation following shutdown of the Davis Besse nuclear plant.
- Developed engineering designs, implemented modifications, and tested control systems at power through successful program recovery.

**Babcock & Wilcox** (March 1970-November 1985)

**Project Engineer, Plant Services** (September 1984-November 1985)
- Developed and deployed hardware and inspection services for repair and maintenance of steam generators and pressure vessels.
- Managed field installation of fuel handling bridge in Kumatori, Japan.

**Project Manager, International Business** (June 1982-September 1984)

- Developed markets for B&W technology services in Europe and the Pacific Basin in partnership with international companies such as Brown Boveri (Germany), Framatome (France), Sumitomo (Japan) and McDermott International (Hong Kong).

**Principal Engineer, Plant Performance** (January 1980-June 1982)

- Supervised 9 member team developing operator guidelines for anticipated reactor transients.
- Specialized in original control system analysis and design, principal accomplishments including:
  - Developed course on plant control systems,
  - Consulted onsite on steam generator performance problems,
  - Completed operational/accident transient analyses for several nuclear contracts,
  - Performed failure modes and effects analysis for the integrated reactor control system,
  - Extended methods for reactor power determination,
  - Developed original analyses and conceptual control schemes for steam generator overfill, water hammer transients, anticipated transients without reactor scram, two-phase natural circulation cooling, and reactor vessel embrittlement.

**Technical Advisor, Plant Design** (January 1976-December 1980)

- On loan to Brown Boveri, Germany, through licensing of the reactor safety systems for the Muehlheim-Kaerlich nuclear plant, to consult on technical licensing issues and oversee the development of complex, nonproprietary computer codes for reactor safety analyses.

**Senior Engineer, Technical Staff** (March 1970-January 1976)

- Applied internal and industry research to nuclear plant design, provided technical assistance to the engineering department, and developed computer codes licensed for performing transient thermal-hydraulic analyses.
- On loan to Duke Power as test engineer during hot functional testing at Oconee nuclear power station.

## EDUCATION

Master of Science/Bachelor of Science, Mechanical Engineering; Lehigh University
Master of Engineering Administration; George Washington University
Bachelor of Applied Accounting; Tennessee Wesley College
Associate of Computer Science; Chattanooga State
Associate of Electronics; U.S. Naval Electronics School

## CERTIFICATIONS

Registered Professional Engineer (#20668, VA); Certified Quality Systems Auditor, ISO-9000 (#Q05630); Certified Manager (#02929); Toastmasters International (Able Toastmaster)

## PROFESSIONAL ASSOCIATIONS

American Society of Mechanical Engineers, American Nuclear Society, American Society for Quality Control, Institute of Electrical and Electronic Engineers

# Richard J. Laufer

**Experience**:

2/99 - Present: NRC Project Manger - Shearon Harris Nuclear Power Plant

Serve as the Headquarters Focal Point for Information and Communication on all issues concerning the Shearon Harris Nuclear Power Plant. Maintain nearly daily communication with the licensee, the resident inspectors, and the regional staff. Participate in all significant licensee meetings in the region and on-site. Serve as Back-up Project Manager (PM) for another plant in the Project Directorate (currently H.B. Robinson).

Prepare and coordinate the numerous documents generated to support the licensing activities of the assigned plant. These documents include license amendments and exemptions and their associated environmental assessments and Federal Register Notice, Task Interface Agreement Responses, controlled correspondence, and numerous letters to the licensee associated with closing out Generic Letters, relief requests, and requests for additional information.

Coordinate, participate, and manage meetings and briefings by ensuring that the appropriate NRC contacts are informed, that meeting notices are prepared, and by preparing an accurate and concise meeting summary in a timely manner.

2/98 - 2/99: NRC Project Manager - Duane Arnold Energy Center

7/93 - 2/98: NRC Project Manager - Kewaunee Nuclear Power Plant

2/93 - 7/93: NRC Project Engineer - Division of Reactor Projects

5/89 - 2/93: NRC Operator Licensing Examiner - Operator Licensing Branch

- Certified NRC Operator Licensing Examiner on Westinghouse pressurized water reactors and non-power reactors

3/86 - 5/89: Engineering Division Officer on Navy nuclear submarine USS Vallejo (SSBN 658) (Qualified as Engineering Officer of the Watch, Engineering Duty Officer)

**Training:**

1/90 Completed NRC's Westinghouse Technology Full Series Course

5/84- 3/86: Navy nuclear power training

**Education**:

5/84: B.S. Degree in Systems Engineering; U. S. Naval Academy, Annapolis, MD

# RESUME

Donald G. Naujock

College Education:
BS, The University of Wisconsin-Milwaukee, 1971, Material Science Engineering
MS, The University of Wisconsin-Milwaukee, 1972, Metallurgical

Experience:
August 1991 to Present: U.S. Nuclear Regulatory Commission, Mail Stop O-7D4, Washington, D.C. 20555

Employed as a technical reviewer for the Materials and Chemical Engineering Branch of the Division of Engineering of the Office of Nuclear Reactor Regulation. My accomplishments are: Assessed the Performance Demonstration Initiative (PDI) program developed by the nuclear utilities for implementing Appendix VIII to Section XI of the American Society of Mechanical Engineers Code. Developed and coordinated the staff's effort to reference Appendix VIII in the final rule that was issued in the *Federal Register* on September 22, 1999. Developed the chemical ranges in Draft Regulatory Guide DG-1070 for verification of product check analysis used in the commercial dedication process of steels. Prepared Information Notice 92-60, "Valve Stem Failure Caused By Embrittlement, " August 20, 1992, and Proposed Generic Communication; "Effectiveness of Ultrasonic Testing Systems in Inservice Inspection Programs," *Federal Register,* Volume 61, No. 252, Pages 69120 - 69124. Reviewed over 30 ASME Code cases for endorsement by the NRC; the Code cases covered subjects, such as, nondestructive testing, nondestructive techniques, inservice inspections, welding, and materials. Reviewed over 50 submittals requesting alternatives to the 10 CFR 50.55a paragraphs (c),(d),(e),(f),(g), and (h); the submittals covered subjects, such as, inspections of reactor vessels using performance demonstrated ultrasonic techniques, use of wire penetrameters in radiography, changes in hydrostatic testing, inspection coverage of welds, inspections of control rod drive welds, ultrasonic testing qualification for intergranular stress-corrosion cracking, ultrasonic testing to determine water level in piping, ultrasonic testing in lieu of radiographic testing of dissimilar metal welds, and weld repairs/overlay of non-structural seal welds. Participated in over 10 vendor inspections to examine issues on securing test specimens using electrical discharge machining, ultrasonic testing for cold cracks at cladding-to-basemetal interface, processing forged material, chemical analysis of commercial grade material, demonstration the phase array ultrasonic testing system, manufacture of small diameter fasteners, process and ultrasonic testing of zirconium alloy fuel rod assemblies, and heat treating of commercial dedicated material. Technical monitor of over 7 contracts with national laboratories on participating in the evaluation of ultrasonic techniques, assisting in reviewing public comments and topical reports, and developing specifications for a mobile nondestructive testing facility,

Co-Authored, "U.S. Nuclear Regulatory Commission Perspective on Performance Demonstration of Ultrasonic Testing Systems," Presented at the 22nd MPA-Seminar, "Safety and Reliability of Plant Technology," October 10 and 11, 1996, University of Stuttgart, Germany.

December 1988 to June 1991, Tennessee Valley Steel Corporation, Chief Metallurgist

My accomplishment was establishing a quality control facility and operation to the restoration of an abandon mini-steel mill. Responsible for all company quality control and customer service functions. Maintained steel tonnage records and both chemical and physical testing labs for mini-steel mill. Trained technicians to test cooling water pH and hardness, to operate optical emission spectrometer, to calculate furnace and ladle alloy additions, and to coordinate plant electrical power demand with local utility. Wrote safe operating procedures for equipment, refurbished test preparation equipment, and participated in accident reviews. Resolved slag entrapment, porosity, chemical variations, and off dimensional billets. Solved customer processing and material selection problems and coordinated shipping, testing, and storage of hazardous furnace emission by-products.

March 1986 to August 1988, Tennessee Valley Authority, Metallurgical Engineer
(Contractor from 3/86 to 8/87 and employee from 10/87 to 8/88)

My accomplishments are establishing the technical justification for selecting Type 347 modified stainless steel for nuclear piping applications, developing heat treatment and welding techniques for enhancing corrosion resistance and reducing manufacturing costs. Resolved material related employee concerns, fabrication inconsistencies, and intergranular stress corrosion cracking issues. Investigated the use of counterfeit fasteners, interfaced with vendors, and upgraded procurement procedures. Provided guidance in the application of ASME Codes and ASTM Specifications for construction personnel.

August 1982 to January 1986, Brush Wellman Inc, Senior Manufacturing Engineer

My accomplishments are a 50 % increase in induction melting capacity and a 50% increase in new electric arc furnace uptime. Coordinated $3.5 million dollar cast shop expansion; justified, selected, debugged, and evaluated performance of new induction furnaces, heat treat furnace, planer mill, band saws, gun drill, and ventilation equipment. Redesigned tap hole configuration, charge material handling bins, and molds for electric arc furnace. Responsible for product quality, productivity, and yield for beryllium-copper alloys processed through electric arc furnaces, induction furnaces, direct chill casting machines, and billet conditioning equipment. Wrote routings, job descriptions, process procedures, operating standards, and equipment specifications. Maintained variances and 5-year expansion plan. Investigated furnace failures and established preventive measures to reduce reoccurrences.

ATTACHMENT 2

LIST OF DOCUMENTS SERVED ON ORANGE COUNTY
IN RESPONSE TO FIRST DISCOVERY REQUESTS TO THE NRC STAFF

Contention 2 - Criticality

1. Letter from Brian Grimes, NRC, to All Power Reactor Licensees, dated April 14, 1978.

2. Letter from Brian Grimes, NRC, to All Power Reactor Licensees, dated January 18, 1979, modifying letter of April 14, 1978.

2. Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Strorage Facility Design Basis," dated September 23, 1981.

Contention 3

1. X-MET, Portable XRF Analyzer brochure.

2. Request for Additional Information, dated December 23, 1998, FAXed copy

3. Request for Additional Information, dated March 24, 1999.

4. Request for Additional Information, dated September 10, 1999, FAXed copy.

5. Request for Additional Information, dated April 21, 1999, FAXed copy.

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorization to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

Div of Op Reactors
Operational Technology

OT POSITION FOR REVIEW AND ACCEPTANCE OF
SPENT FUEL STORAGE AND HANDLING APPLICATIONS

## I. BACKGROUND

Prior to 1975, low density spent fuel storage racks were designed with a large pitch, to prevent fuel pool criticality even if the pool contained the highest enrichment uranium in the light water reactor fuel assemblies. Due to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better utilize available space. In the case of operating plants the new rack system interfaces with the old fuel pool structure. A proposal for installation of high density storage racks may involve a plant in the licensing stage or an operating plant. The requirements of this position do not apply to spent fuel storage and handling facilities away from the nuclear reactor complex.

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, an environmental impact statement or environmental impact appraisal shall be prepared in which five specific factors in addition to the normal cost/benefit balance and environmental stresses should be applied, balanced and weighed.

The overall design objectives of a fuel storage facility at the reactor complex are governed by various Regulatory Guides, the Standard Review Plan, and industry standards which are listed in the reference section. Based on the reviews of such applications to date it is obvious that the staff had to request additional information that could be easily included in an adequately documented initial submittal. It is the intent of this document to provide guidance for the type and extent of information needed to perform the review, and to indicate the acceptance criteria where applicable.

## II. REVIEW DISCIPLINES

The objective of the staff review is to prepare (1) Safety Evaluation Report, and (2) Environmental Impact Appraisal. The broad staff disciplines involved are nuclear, mechanical, material, structural, and environmental.

Nuclear and thermal-hydraulic aspects of the review include the potential for inadvertant criticality in the normal storage and handling of the spent fuel, and the consequences of credible accidents with respect to criticality and the ability of the heat removal system to maintain sufficient cooling.

Mechanical, material and structural aspects of the review concern the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand the effects of natural phenomena such as earthquakes, tornadoes, flood, effects of external and internal missiles, thermal loading, and also other service loading conditions.

The environmental aspects of the review concern the increased thermal and radiological releases from the facility under normal as well as accident conditions, the occupational radiation exposures, the generation of radioactive waste, the need for expansion, the commitment of material and nonmaterial resources, realistic accidents, alternatives to the proposed action and the cost-benefit balance.

The information related to nuclear and thermal-hydraulic type of analyses is discussed in Section III.

The mechanical, material, and structural related aspects of information are discussed in Section IV.

The information required to complete an environmental impact assessment, including the five factors specified by the Commission, is provided in Section V.

## III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

1. Neutron Multiplication Factor

To include all credible conditions, the licensee shall calculate the effective neutron multiplication factor, $k_{eff}$, in the fuel storage pool under the following sets of assumed conditions:

1.1 Normal Storage

a. The racks shall be designed to contain the most reactive fuel authorized to be stored in the facility without any control rods or any noncontained* burnable poison and the fuel shall be assumed to be at the most reactive point in its life.

b. The moderator shall be assumed to be pure water at the temperature within the fuel pool limits which yields the largest reactivity.

c. The array shall be assumed to be infinite in lateral extent or to be surrounded by an infinitely thick water reflector and thick concrete,** as appropriate to the design.

d. Mechanical uncertainties may be treated by assuming "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

e. Credit may be taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption, provided a means of inspection is established (refer to Section 1.5).

1.2 Postulated Accidents

The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies. The

*"Noncontained" burnable poison is that which is not an integral part of the fuel assembly.

**It should be noted that under certain conditions concrete may be a more effective reflector than water.

postulated accidents shall include: (1) dropping of a fuel element on top of the racks and any other achievable abnormal location of a fuel assembly in the pool; (2) a dropping or tipping of the fuel cask or other heavy objects into the fuel pool; (3) effect of tornado or earthquake on the deformation and relative position of the fuel racks; and (4) loss of all cooling systems or flow under the accident conditions, unless the cooling system is single failure proof.

### 1.3 Calculation Methods

The calculation method and cross-section values shall be verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. Sufficiently diverse configurations shall be calculated to render improbable the "cancellation of error" in the calculations. So far as practicable the ability to correctly account for heterogeneities (e.g., thin slabs of absorber between storage locations) shall be demonstrated.

A calculational bias, including the effect of wide spacing between assemblies shall be determined from the comparison between calculation and experiment. A calculation uncertainity shall be determined such that the true multiplication factor will be less than the calculated value with a 95 percent probability at a 95 percent confidence level. The total uncertainity factor on $k_{eff}$ shall be obtained by a statistical combination of the calculational and mechanical uncertainties. The $k_{eff}$ value for the racks shall be obtained by summing the calculated value, the calculational bias, and the total uncertainty.

### 1.4 Rack Modification

For modification to existing racks in operating reactors, the following information should be provided in order to expedite the review:

(a) The overall size of the fuel assembly which is to be stored in the racks and the fraction of the total cell area which represents the overall fuel assembly in the model of the nominal storage lattice cell;

(b) For $H_2O$ + stainless steel flux trap lattices; the nominal thickness and type of stainless steel used in the storage racks and the thermal (.025 ev) macroscopic neutron absorption cross section that is used in the calculation method for this stainless steel;

(c) Also, for the $H_2O$ + stainless steel flux trap lattices, the change of the calculated neutron multiplication factor of infinitely long fuel assemblies in infinitely large arrays in the storage rack (i.e., the $k_\infty$ of the nominal fuel storage lattice cell and the changed $k_\infty$) for:

(1) A change in fuel loading in grams of $U^{235}$, or equivalent, per axial centimeter of fuel assembly where it is assumed that this change is made by increasing the enrichment of the $U^{235}$; and,

(2) A change in the thickness of stainless steel in the storage racks assuming that a decrease in stainless steel thickness is taken up by an increase in water thickness and vice versa;

(d) For lattices which use boron or other strong neutron absorbers provide:

(1) The effective areal density of the boron-ten atoms (i.e., $B^{10}$ atoms/cm$^2$ or the equivalent number of boron-ten atoms for other neutron absorbers) between fuel assemblies.

(2) Similar to Item C, above, provide the sensitivity of the storage lattice cell $k_\infty$ to:

(a) The fuel loading in grams of $U^{235}$, or equivalent, per axial centimeter of fuel assembly,

(b) The storage lattice pitch; and,

(c) The areal density of the boron-ten atoms between fuel assemblies.

## 1.5 Acceptance Criteria for Criticality

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions

(1) For those facilities which employ a strong neutron absorbing material to reduce the neutron multiplication factor for the storage pool, the licensee shall provide the description of onsite tests which will be performed to confirm the presence and retention of the strong absorber in the racks. The results of an initial, onsite verification test shall show within 95 percent confidence limits that there is a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95. In addition, coupon or other type of surveillance testing shall be performed on a statistically acceptable sample size on a

periodic basis throughout the life of the racks to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95.

(2) Decay Heat Calculations for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 75/087).

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

Conservative methods should be used to calculate the maximum fuel temperature and the increase in temperature of the water in the pool. The maximum void fraction in the fuel assembly and between fuel assemblies should also be calculated.

Ordinarily, in order not to exceed the design heat load for the spent fuel cooling system it will be necessary to do a certain amount of cooling in the reactor vessel after reactor shutdown prior to moving fuel assemblies into the spent fuel pool. The bases for the analyses should include the established cooling times for both the usual refueling case and the full core off load case.

A potential for a large increase in the reactivity in an $H_2O$ flux trap storage lattice exists if, somehow, the water is kept out or forced out of the space between the fuel assemblies, conceivably by trapped air or steam. For this reason, it is necessary to show that the design of the storage rack is such that this will not occur and that these spaces will always have water in them. Also, in some cases, direct gamma heating of the fuel storage cell walls and of the intercell water may be significant. It is necessary to consider direct gamma heating of the fuel storage cell walls and of the intercell water to show that boiling will not occur in the water channels between the fuel assemblies. Under postulated accident conditions where all non-Category I spent fuel pool cooling systems become inoperative, it is necessary to show that there is an alternate method for cooling the spent pool water. When this alternative method requires the installation of alternate components or significant physical alteration of the cooling system, the detailed steps shall be described, along with the time required for each. Also, the average amount of water in the fuel pool and the expected heat up rate of this water assuming loss of all cooling systems shall be specified.

(4) Potential Fuel and Rack Handling Accidents

The method for moving the racks to and from and into and out of the fuel pool, should be described. Also, for plants where the spent fuel pool modification requires different fuel handling procedures than that described in the Final Safety Analysis Report, the differences should be discussed. If potential fuel and rack handling accidents occur, the neutron multiplication factor in the fuel pool shall not exceed 0.95. These postulated accidents shall not be the cause of the loss of cooling for either the spent fuel or the reactor.

(5) Technical Specifications

To insure against criticality, the following technical specifications are needed on fuel storage in high density racks:

1. The neutron multiplication factor in the fuel pool shall be less than or equal to 0.95 at all times.

2. The fuel loading (i.e., grams of uranium-235, or equivalent, per axial centimeter of assembly) in fuel assemblies that are to be loaded into the high density racks should be limited. The number of grams of uranium-235, or equivalent, put in the plant's technical specifications shall preclude criticality in the fuel pool.

Excessive pool water temperatures may lead to excessive loss of water due to evaporation and/or cause fogging. Analyses of thermal load should consider loss of all pool cooling systems. To avoid exceeding the specified spent fuel pool temperatures, consideration shall be given to incorporating a technical specification limit on the pool water temperature that would resolve the concerns described above. For limiting values of pool water temperatures refer to ANSI-N210-1976 entitled, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," except that the requirements of the Section 9.1.3.III.1.d of the Standard Review Plan is applicable for the maximum heat load with normal cooling systems in operation.

# IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool and the racks. The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed under the environmental discipline. Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. Integrity of the racks and the fuel pool due to a postulated fuel handling accident is reviewed under the mechanical, material, and structural disciplines. Sketches and sufficient details of the fuel handling system should be provided to facilitate this review.

(2) Applicable Codes, Standards and Specifications

Construction materials should conform to Section III, Subsection NF of the ASME* Code. All Materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for stainless steel base metal may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code.

Other materials, design procedures, and fabrication techniques will be reviewed on a case by case basis.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61 respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guide 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.

**American Institute of Steel Construction, Latest Edition.

The peak response from each direction should be combined by square root of the sum of the squares. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

The effect of submergence of the rack system on the damping and the mass of the fuel racks has been under study by the NRC. Submergence in water may introduce damping from two sources, (a) viscous drag, and (b) radiation of energy away from the submerged body in those cases where the confining boundaries are far enough away to prevent reflection of waves at the boundaries. Viscous damping is generally negligible. Based upon the findings of this current study for a typical high density rack configuration, wave reflections occur at the boundaries so that no additional damping should be taken into account.

A report on the NRC study is to be published shortly under the title "Effective Mass and Damping of Submerged Structures (UCRL-52342)," by R. G. Dong. The recommendations provided in this report on the added mass effect provide an acceptable basis for the staff review. Increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described in the report: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to the applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and base slab. Temperature gradient across the rack structure due to differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of Section 3.8.4-II.3 of the Standard Review Plan.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach, two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and,

(b) A static analysis of the rack system by subjecting it to the maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

| Load Combination<br>Elastic Analysis | Acceptance Limit |
|---|---|
| D + L | Normal limits of NF 3231.1a |
| D + L + E | Normal limits of NF 3231.1a |
| D + L + To | 1.5 times normal limits or the lesser of 2 Sy and Su |
| D + L + To + E | 1.5 times normal limits or the leser of 2 Sy and Su |
| D + L + Ta + E | 1.6 times normal limits or the lesser of 2 Sy or Su |
| D + L + Ta + E[1] | Faulted condition limits of NF 3231.1c |
| **Limit Analysis** | |
| 1.7 (D + L) | Limits of XVII-4000 of Appendix XVII of ASME Code Section III |
| 1.7 (D + L + E) | |
| 1.3 (D + L + To) | |
| 1.3 (D + L + E + To) | |
| 1.1 (D + L + Ta + E) | |

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be ammended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

the construction phase should be provided. Methods for structural qualification of special poison materials utilized to absorb neutron radiation should be described. The material for the fuel rack is reviewed for compatibility inside the fuel pool environment. The quality of the fuel pool water in terms of the pH value and the available chlorides, fluorides, boron, heavy metals should be indicated so that the long-term integrity of the rack structure, fuel assembly, and the pool liner can be evaluated.

Acceptance criteria for special materials such as poison materials should be based upon the results of the qualification program supported by test data and/or analytical procedures.

If connections between the rack and the pool liner are made by welding, the welder as well as the welding procedure for the welding assembly shall be qualified in accordance with the applicable code.

If precipitation hardened stainless steel material is used for the construction of the spent fuel pool racks, hardness testing should be performed on each rack component of the subject material to verify that each part is heat treated properly. In addition, the surface film resulting from the heat treatment should be removed from each piece to assure adequate corrosion resistance.

(8) Testing and Inservice Surveillance

Methods for verification of long-term material stability and mechanical integrity of special poison material utilized for neutron absorption should include actual tests.

Inservice surveillance requirements for the fuel racks and the poison material, if applicable, are dependent on specific design features. These features will be reviewed on a case by case basis to determine the type and the extent of inservice surveillance necessary to assure long-term safety and integrity of the pool and the fuel rack system.

V. COST/BENEFIT ASSESSMENT

1. Following is a list of information needed for the environmental Cost/Benefit Assessment:

1.1 What are the specific needs that require increased storage capacity in the spent fuel pool (SFP)? Include in the response:

(a) status of contractual arrangements, if any, with fuel-storage or fuel-reprocessing facilities,

(b) proposed refueling schedule, including the expected number of fuel assemblies that will be transferred into the SFP at each refueling until the total existing capacity is reached,

(c) number of spent fuel assemblies presently stored in the SFP,

(d) control rod assemblies or other components stored in the SFP, and

(e) the additional time period that spent fuel assemblies would be stored onsite as a result of the proposed expansion, and

(f) the estimated date that the SFP will be filled with the proposed increase in storage capacity.

1.2 Discuss the total construction associated with the proposed modification, including engineering, capital costs (direct and indirect) and allowances for funds used during construction.

1.3 Discuss the alternative to increasing the storage capacity of the SFP. The alternatives considered should include:

(a) shipment to a fuel reprocessing facility (if available),

(b) shipment to an independent spent fuel storage facility,

(c) shipment to another reactor site,

(d) shutting down the reactor.

The discussion of options (a), (b) and (c) should include a cost comparison in terms of dollars per KgU stored or cost per assembly. The discussion of (d) should include the cost for providing replacement power either from within or outside the licensee's generating system.

1.4 Discuss whether the commitment of material resources (e.g., stainless steel, boral, $B_4C$, etc.) would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. Describe the material resources that would be consumed by the proposed modification.

1.5 Discuss the additional heat load and the anticipated maximum temperature of water in the SFP which would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems and whether there will be any significant increase in the amount of heat released to the environment.

## V.2. RADIOLOGICAL EVALUATION

2. Following is a list of information needed for radiological evaluation:

2.1 The present annual quantity of solid radioactive wastes generated by the SFP purification system. Discuss the expected increase in solid wastes which will result from the expansion of the capacity of the SFP.

2.2. Data regarding krypton-85 measured from the fuel building ventilation system by year for the last two years. If data are not available from the fuel building ventilation system, provide this data for the ventilation release which includes this system.

2.3 The increases in the doses to personnel from radionuclide concentrations in the SFP due to the expansion of the capacity of the SFP, including the following:

(a) Provide a table showing the most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations.

(b) The models used to determine the external dose equivalent rate from these radionuclides. Consider the dose equivalent rate at some distance above the center and edge of the pool respectively. (Use relevant experience if necessary).

(c) A table of recent analysis performed to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

(d) The model and assumptions used to determine the increase, if any, in dose rate from the radionuclides identified in (c) above in the SFP area and at the site boundary.

(e) An estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter media.

(f) The buildup of crud (e.g., $^{58}Co$, $^{60}Co$) along the sides of the pool and the removal methods that will be used to reduce radiation levels at the pool edge to as low as reasonably achievable.

(g) The expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (e) and (f) above.

A discussion of the radiation protection program as it affects (a) through (g) should be provided.

2.4 Indicate the weight of the present spent fuel racks that will be removed from the SFP due to the modification and discuss what will be done with these racks.

V.3 ACCIDENT EVALUATION

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

(b) evaluation of the overhead handling system with respect to Regulatory Guide 1.104.

3.2 If the accident aspects of review do not establish acceptability with respect to either (a) or (b) above, then technical specifications may be required that prohibit cask movement in the spent fuel building.

3.3 If the accident review does not establish acceptability with respect to (b) above, then technical specifications may be required that:

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

(2) indicate the minimum age of fuel in pool sections during movement of heavy loads near the pool. In special cases evaluation of consequences-limiting engineered safety features such as isolation systems and filter systems may be required.

3.4 If the cask drop/tip analysis as in 3.1(a) above is promised for future submittal, the staff evaluation will include a conclusion on the feasibility of a specification of minimum age of fuel based on previous evaluations.

3.5 The maximum weight of loads which may be transported over spent fuel may not be substantially in excess of that of a single fuel assembly. A technical specification will be required to this effect.

3.6 Conclusions that determination of previous Safety Evaluation Reports and Final Environmental Statements have not changed significantly or impacts are not significant are made so that a negative declaration with an Environmental Impact Appraisal (rather than a Draft and Final Environmental Statement) can be issued. This will involve checking realistic as well as conservative accident analyses.

## VI. REFERENCES

1. Regulatory Guides

   1.13 - Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

   1.29 - Seismic Design Classification

   1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

   1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

   1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

   1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section III, Division 1

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

   4. American Society of Civil Engineers, Suggested Specification for Structures of Aluminium Alloys 6061-T6 and 6067-T6

5. The Aluminium Association, Specification for Aluminium Structures

GL79004

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20565

January 18, 1979

To All Power Reactor Licensees

Gentlemen:

Our letter of April 14, 1978, provided NRC Guidance entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications." Enclosed are modifications to this document for your information and use. These involve pages IV-5 and IV-6 of the document and comprise modified rationale and corrections.

Sincerely,

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosure:
Pages IV-5 and IV-6

cc w/enclosure:
Service List

7903080173

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described in "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below. When buckling loads are considered in the design, the structural acceptance criteria shall be limited by the requirements of Appendix XVII-2110(b) of the ASME Boiler and Pressure Vessel Code.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

IV-5

TABLE

| Load Combination<br>Elastic Analysis | Acceptance Limit |
|---|---|
| D + L | Normal limits of NF 3231.1a |
| D + L + E | Normal limits of NF 3231.1a |
| D + L + To | Lesser of 2Sy or Su stress range |
| D + L + To + E | Lesser of 2Sy or Su stress range |
| D + L + Ta + E | Lesser of 2Sy or Su stress range |
| D + L + Ta + E1 | Faulted condition limits of NF 3231.1c |
| Limit Analysis | |
| 1.7 (D + L) | Limits of XVII-4000 of Appendix XVII of ASME Code Section III |
| 1.7 (D + L + E) | |
| 1.3 (D + L + To) | |
| 1.3 (D + L + E + To) | |
| 1.1 (D + L + Ta + E) | |

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be amended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

IV-6

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 23 1981

MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safety

FROM: Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: DRAFT 1, REGULATORY GUIDE 1.13, REVISION 2,
"SPENT FUEL STORAGE FACILITY DESIGN BASIS"

Enclosed for initial review of the ACRS Regulatory Activities Subcommittee are 20 copies of Revision 2 to Regulatory Guide 1.13 (Enclosure 1) and 20 copies of the Draft Value/Impact Assessment (Enclosure 2).

The draft regulatory guide is a proposed revision to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," which is being revised to endorse ANSI N210-1976/ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

The draft regulatory guide, which was originally scheduled for review at the September 9th meeting, was withdrawn to insure the incorporation of all necessary input from Division Offices.

Since this draft is preliminary, additional staff efforts, including review and resolution of public comments, will be necessary prior to implementation of a regulatory position. ACRS Regulatory Activities Subcommittee comments and recommendations are requested on the proposed regulatory position.

Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

cc: Public Document Room

Enclosures: as stated

# DRAFT 1 OF REVISION 2 TO REGULATORY GUIDE 1.13
# SPENT FUEL STORAGE FACILITY DESIGN BASIS

## A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing this criterion.

## B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard which details minimum design requirements for 10 CFR Part 50 light water reactor spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor

Spent Fuel Storage Facilities at Nuclear Power Stations" by the American National Standards Institute on April 12, 1976.

These facilities must be designed to:

a. Prevent loss of water from the fuel pool that would uncover fuel.

b. Protect the spent fuel from mechanical damage.

c. Provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. Loss of Water from Storage Pool

Unless protective measures are taken, loss of water from a fuel storage pool could cause overheating of the spent fuel, resultant damage to fuel cladding integrity, and could result in a release of radioactive materials to the environment. Natural events, such as earthquakes or high winds, could damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity would alleviate these concerns.

Dropping of heavy loads, such as a 100-ton fuel cask, although of low probability, should be considered in plant arrangements where such loads are positioned or moved in or over the spent fuel pool. Cranes which are capable of carrying heavy loads should be prevented, preferably by design rather than interlocks, from moving into the vicinity of the pool.

The negative pressure in the fuel handling building during movement of spent fuel should be at least minus 3.2 mm (-0.125 inches) water gauge to prevent exfiltration and to assure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment.

Even if the measures described above which are used to maintain the desired negative pressure are followed, small leaks from the building may still occur as a result of structural failure or other unforeseen events. For example, equipment failures in systems connected to the pool could result in loss of water from the pool if this loss is not prevented by design. A permanent fuel-pool-coolant makeup system with a moderate capability, and with suitable redundancy or backup, could prevent the fuel from being uncovered if these leaks should occur. Early detection of pool leakage and fuel damage could be provided by both pool-water-level monitors and radiation monitors. Both types of monitors should be designed to alarm both locally and in a continuously manned location. Timely operation of building filtration systems can be assured if these systems are actuated by a signal from local radiation monitors.

2. Mechanical Damage to Fuel

The release of radioactive material from fuel may occur during the refueling process, and at other times, as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or the dropping of objects onto fuel elements.

Missiles generated by high winds are also a potential cause of mechanical damage to fuel. This concern could be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. Limiting Potential Offsite Exposures

A relatively small amount of mechanical damage to the fuel or fuel overheating might cause significant offsite doses of radiation if no dose reduction features are provided. Use of a controlled leakage building surrounding the fuel storage pool, with associated capability to limit releases of radioactive material resulting from a refueling accident, would appear feasible and do much to eliminate this concern.

For the spent fuel pool cooling, makeup and cleanup systems, the staff will consider the design acceptable if it includes seismic Category 1 and tornado protection for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the storage building's filtration-ventilation systems. The pool building's filtration-ventilation systems should be designed to meet the guidelines of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

In all activities involving personnel exposure to radiation, attention should be directed toward keeping occupational radiation as low as reasonably achievable (ALARA). Efforts toward maintaining exposures ALARA should be included in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

## C. REGULATORY POSITION

The requirements that are included in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations"[1] are generally acceptable to the NRC staff. The staff has determined that this standard provides an adequate basis for complying with the requirements of General Design Criterion 61 "Fuel Storage and Handling and Radioactivity Control" of Appendix A "General Design Criteria for Nuclear Power Plants" to 10 CFR Part 50 as related to light water reactors and subject to the following clarifications and modifications:

1. The example in Section 4.2.4.3(1) should be modified. The inventory of radioactive materials that could possibly leak from the spent fuel building should correspond to the amount predicted to leak under the postulated maximum damage conditions resulting from the dropping of a spent fuel assembly in the spent fuel building. However, in any event, the inventory should not be less than the amount available due to rupture of all fuel rods of a spent fuel assembly. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."[2]

2. In addition to meeting the requirements of Section 5.1.12 the maximum potential kinetic energy capable of being developed by those objects handled

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[1]Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

[2]Copies of Regulatory Guides may be obtained from the U.S. Nuclear Regulatory Commisson, Washington, D.C. 20555.

above stored spent fuel, if dropped, is not to exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

3. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

4. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to keep tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to keep missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado missiles or from seismic damage to assure that nothing bypasses the ESF grade filtration system in the containment building. ~~In the event an earthquake or a tornado missile damages both the fuel pool containment and the fuel pool cooling system, no credit can be given to the filtration system used to reduce the amount of airborne radioactivity~~.

5. In addition to meeting the requirements of Section 5.1.5.3, provisions should be made for handling highly radioactive non-fuel ~~irradiated~~ components in fuel pools. Either the design of the retrieval system or administrative controls should be included which would prohibit unknowing retrieval of irradiated components.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface between the cask venting system and the ~~installed~~ building ventilation system should be provided. This interface would provide for the proper handling of the "vent-gas" generated from filling a dry, loaded cask with water and thereby minimizing personnel exposure from the untreated off gas.

7. In order to limit the potential offsite release of radioactivity during a Condition IV fuel handling accident, Section 5.3.3 should include the requirement that the released radioactivity be either contained or removed by filtration so that the dose to an individual is less than 10 CFR Part 100 guidelines. The calculated offsite dose to an individual from such an event should be well within (approximately 25% of) the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to assure that released activity does not bypass the filtration system, the engineered safety feature fuel storage building ventilation should provide and maintain a negative pressure of at least ~~minus~~ 3.2mm (0.125 inches), water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed such that travel directly over the spent fuel storage pool or safety-related equipment is not possible. This should be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I and safety-related structures and equipment should be subject to a quality assurance program which meets the applicable provisions of Appendix B to 10 CFR Part 50. Further, those programs should obtain guidance from Regulatory Guide 1.28 endorsing ANSI N45.2 "Quality Assurance Program Requirements for Nuclear Facilities" and the applicable provisions of ANSI N45.2 daughter standards endorsed by Regulatory Guides.

The Regulatory Guides endorsing the applicable ANSI N45.2 daughter standards are as follows:

| | |
|---|---|
| 1.30 | Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (N45.2.4). |
| 1.38 | Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (N45.2.2). |
| 1.58 | Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (N45.2.6). |
| 1.64 | Quality Assurance Requirements for the Design of Nuclear Power Plants (N45.2.11). |
| 1.74 | Quality Assurance Terms and Definitions (N45.2.10). |
| 1.88 | Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (N45.2.9). |
| 1.94 | Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (N45.2.5). |

| | |
|---|---|
| 1.116 | Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (N45.2.8). |
| 1.123 | Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (N45.2.13). |

10. The spent fuel pool water temperature of 65.6°C (150°F) stated in Section 6.6.1(2)(a) exceeds the NRC staff recommended limit. With the normal cooling system in operation, the pool water temperature should be kept at or below 60°C (140°F) with full core offload except when the pool water temperature is based on comparative analyses of the pool conditions that have been found acceptable previously. The spent fuel pool water temperature recommended limits for normal and abnormal cases are indicated in the table below.

NORMAL OPERATION

| Case I | Case II |
|---|---|
| . both trains operational | . both trains operational |
| . normal refueling | . full core offload |
| . pool full of spent fuel | . pool full of spent fuel |
| Maximum operating temperature | Maximum operating temperature |
| < 48.9°C (120 °F) | < 60°C (140° F) |
| based on fogging criteria and personnel comfort | to protect the ion exchange resin from degradation |

ABNORMAL OPERATION

| Case III | Case IV |
|---|---|
| . one train operational | . no cooling loops operational |
| . normal refueling | . full core offload |
| . pool full of spent fuel | . pool full of spent fuel |
| Maximum operating temperature <60°C (140°F) | Pool boiling permitted |

11. A nuclear criticality safety analysis should be performed in accordance with Annex A for each light water reactor spent fuel storage facility that involves the handling, transfer, or storage of spent fuel assemblies.

12. Sections 6.4 and 9 of ANS 57.2 lists codes and standards that are referenced in this standard. Endorsement of ANS 57.2 by this regulatory guide does not constitute an endorsement of the referenced codes and standards.

## D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This guide reflects current NRC staff practice for construction permit review. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission regulations, the methods described herein will be used in the evaluation of license applications docketed after ______________.

Draft - 9/22/81-clf
CSchulten- Job A#1

# appendix ~~ANNEX~~ A

## Nuclear Criticality Safety

### 1. Scope of Nuclear Criticality Safety Assessment

1.1 A nuclear criticality safety analysis shall be performed for each light water reactor spent fuel storage facility system that involves the handling, transfer, or storage of spent fuel assemblies.

1.2 The nuclear criticality safety analysis shall demonstrate that each reactor spent fuel storage facility system is subcritical ($k_{eff}$ shall not exceed 0.95).

1.3 The nuclear criticality safety analysis shall include consideration of all credible normal and abnormal operating occurrences, including:

a) Accidental tipping or falling of a spent fuel assembly

b) Accidental tipping or falling of a storage rack during transfer

c) Misplacement of a spent fuel assembly

d) Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system.

e) Fuel drop accidents

f) Stuck fuel assembly/crane uplifting forces

g) Horizontal motion of fuel before complete removal from rack

h) Placing a fuel assembly along the outside of rack

i) Objects that may fall onto the stored spent fuel assemblies

1.4 At all locations in the reactor spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis shall demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

1.5 The nuclear criticality safety analysis shall explicitly identify spent fuel assembly characteristics upon which subcriticality in the reactor spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis shall explicitly identify design limits upon which subcriticality depends that require physical verfication at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis shall explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

## 2. Calculational Methods and Codes

Methods used to calculate subcriticality shall be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety." (Endorses ANSI N16.9-1975)

Draft - 9/22/81-clf
CSchulten- Job A#1

## 3. Method to Establish Subcriticality

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks under normal and credible abnormal conditions shall be equal to or less than an established maximum allowable multiplication factor $k_a$; i.e.,

$$k_s \leq k_a \qquad \text{(Eq. 1)}$$

where

$k_s$ = the evaluated maximum multiplicaton factor of fuel in the spent fuel storage racks, including any necessary allowance for statistical uncertainties in the calculational technique such as in Monte Carlo calculations.

The maximum allowable multiplication factor shall be calculated from the expression:

$$k_a = k_c - \Delta k_u - \Delta k_m \qquad \text{(Eq. 2)}$$

where

$k_c$ = $k_{eff}$ computed for the most reactive fuel assembly at the most reactive point by the same calculational method which was used ~~for the benchmark experiments.~~

<u>Note</u>: $k_c$ is the value of $k_{eff}$ that results from the calculation of the benchmark experiments using a particular calculational method. The value represents a combination of theoretical technique and numerical data. (For more detail, see Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety.")

~~$\Delta k_u$ = The uncertainty in the benchmark experiments.~~

~~$\Delta k_m$ = The value required to assure an accepted margin of subcriticality.~~

3.2 $\Delta k_u$ shall include both uncertainties in the benchmark experiments as well as uncertainties in the bias which result from extrapolation of the benchmark experiments into the range of parameters encountered in the spent fuel storage rack design.

3.3 $\Delta k_m$ shall provide an adequate margin of subcriticality under the operating limitations and Design Events I through IV, and shall be no less than 0.02 (new fuel when stored dry).*

3.4 In the absence of information that justifies a smaller margin of subcriticality, value of 0.05 shall be assumed for $\Delta k_m$ for the design of spent fuel storage racks (spent fuel).

## 4. Storage Rack Analysis Assumptions

4.1 [~~The-fuel-assembly-assumed-for-storage-facility-design-shall-be-one of-the-following:~~] The spent fuel storage rack module design shall be based on one of the following assumptions for the fuel:

a) the most reactive fuel assembly to be stored at the most reactive point in the assembly life [~~with-no-allowance-for fission-product-content-due-to-burn-up~~]; or

---

*Additions shown by underline and a vertical line in each margin. Deletions shown by brackets and crossouts.

b) the most reactive fuel assembly to be stored based on a minimum confirmed burn up. [~~If credit is taken for burnup, an allowable fuel assembly reactivity shall be established and it shall be shown by actual measurement that each fuel assembly meets this criterion before it is allowed to be placed in storage.~~] (See Annex B.)

Both types of rack modules may be present in the same storage pool.

4.2 Determination of the most reactive spent fuel assembly shall include consideration of the following parameters:

. maximum fissile fuel loading,
. fuel rod diameter,
. fuel rod cladding material and thickness,
. fuel pellet density,
. fuel rod pitch and total number of fuel rods within assembly,
. absence of fuel rods in certain locations, and
. burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design shall be the arrangement that results in the highest value of $k_s$ considering:

a) spacing between assemblies,
b) moderation between assemblies, and
c) fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of $k_s$ shall include consideration of the following:

a) eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,

b) dimensional tolerances,

c) construction materials,

d) fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e) presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f) presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Determination of burn up for storage shall be made in racks for which credit is taken for burn up. The following methods are acceptable:

a) a minimum allowed fuel assembly reactivity shall be established and a reactivity measurement shall be performed to assure that each assembly meets this criterion; or

b) a minimum fuel assembly burn up value shall be established as determined by initial fuel assembly enrichment or other correlative parameters and a measurement shall be performed to assure each fuel assembly meets the established criterion; or

c) a minimum fuel assembly burn up value shall be established as determined by initial fuel assembly enrichment or other correlative parameters and an analysis of each fuel assembly's exposure history shall be performed to determine its burn up. The analyses shall be performed under strict administrative control using approved written procedures. The procedures shall provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in Section 1.4.

The uncertainties in determining fuel assembly storage acceptance criteria shall be considered in establishing storage rack reactivity, and auditable records shall be kept of the method used to determine fuel assembly storage acceptance criterion for as long as the fuel assemblies are stored in the racks.

Consideration shall be given to the axial distribution of burn up in the fuel assembly and a limit shall be set on the length of the fuel assembly which is permitted to have a lower average burn up than the fuel assembly average.

5. Use of Neutron Absorbers in Storage Rack Design

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a) The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b) Fixed neutron absorbers shall be an integral, non-removable part of the storage rack.

c) When a fixed neutron absorber is used as the primary nuclear criticality safety control, there shall be provision to:

1) initially confirm absorber presence in the storage rack, and

2) periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water shall not normally be used in the evaluation of $k_s$. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

# ANNEX B

## Most Reactive Fuel Assembly to be Stored Based on a Minimum Confirmed Burnup

If credit is to be taken for fuel burnup in the design of spent fuel storage racks, an acceptable basis for setting and meeting the limit must be established. The rationale for this basis will evolve from many rather complex considerations.

Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, $^{235}U$ depletion, amount of burnable poison, plutonium buildin and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildin are not necessarily the same.

Consideration should be given to how burnup limits are selected and specified for a particular fuel type:

The allowable $^{235}U$ depletion in the spent fuels without burnable poison must not be set too high. If too much depletion is credited in the analysis compared to the range of $^{235}U$ depletion in spent fuel assemblies to be stored, the design could be nonconservative from the standpoint of criticality safety. On the other hand, if too little depletion is credited in the analysis compared to the spent fuel to be stored, then the design will be conservative. Thus a <u>maximum</u> depletion to be allowed in <u>design</u>

can be established consistent with the range of $^{235}U$ depletions expected in the spent fuel assemblies to be stored. (This limit would then correspond to the minimum depletion that would be allowed in a particular fuel assembly type destined to be stored in the racks.)

The allowable plutonium content in the spent fuel upon which design would be based must not be set too low. If design is based on too little plutonium compared to the range of plutonium concentrations that may be in the spent fuel assemblies to be stored in the racks, the design could be non-conservative from the standpoint of nuclear criticality safety. On the other hand, if too much plutonium is credited in the analysis of the storage racks compared to the spent fuel assemblies to be stored, then the design would be conservative. Thus, a minimum plutonium content to be allowed in design can be established consistent with the range of plutonium concentrations expected in the spent fuel assemblies to be stored.

(This limit would then correspond to the maximum plutonium content that would be allowed in a particular fuel assembly type destined to be stored in the racks.)

Credit for fission product content presents special problems, such as the identities and quantities of the various fission products present and how to evaluate the effect of decay rates on the credit taken. The allowable fission product content in the spent fuel upon which design would be based must not be set too high. If design is based on too high of a fission product content compared to the range of fission product concentrations that may be in the spent fuel assemblies to be stored in the racks, the design could be non-conservative from the standpoint of criticality safety. On the other hand, if too few fission products are credited in the analysis of the racks compared to the spent fuel assemblies to be stored, then the design would be conservative. Thus, with proper consideration a maximum fission product content to be allowed in design could be established consistent with the range of fission product concentrations expected in the spent fuel to be stored.

(This limit would then correspond to the minimum fission product content that would be allowed in a particular fuel assembly type to be stored in the racks.)

Finally, consideration should be given to the practical implementation of the spent fuel screening process. Factors to be considered in choosing the screening method should include: [~~Depletion of $^{235}U$ and plutonium and fission product buildin cannot be easily or practically determined analytically. An obvious approach would be to translate the allowable burnup to a net allowable fuel assembly reactivity and then measure every fuel assembly to confirm that the minimum criterion is met.~~]

- accuracy of the method in determining the storage rack reactivity;
- reproducibility of the result, i.e., what is the confidence in the result?
- simplicity of the procedure; i.e., how much disturbance to other operations is involved?;
- accountability, i.e., ease and completeness of recordkeeping; and
- auditability.

# VALUE/IMPACT ASSESSMENT ON NUCLEAR POWER PLANT SPENT FUEL STORAGE FACILITY DESIGN

## 1. PROPOSED ACTION

### 1.1 Description

Each nuclear power plant has a spent fuel storage facility. General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to assure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

### 1.2 Need for Proposed Action

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested this guide be updated.

## 1.3 Value/Impact of Proposed Action

### 1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit application. Therefore, there should be a minimum of cases where the applicant and the staff radically disagree on the design criteria.

### 1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

### 1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

### 1.3.4 Public

No major impact on the public can be foreseen.

## 1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

# 2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also recommendations made by Task A-36 which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" would also be included.

## 3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

## 4. STATUTORY CONSIDERATIONS

### 4.1 NRC AUTHORITY

This guide would fall under the authority and safety requirements of the Atomic Energy Act of 1954, as amended. In particular under General Design Criterion 61, Appendix A, 10 CFR Part 50 of the NRC's implementing regulations.

### 4.2 Need for NEPA Assessment

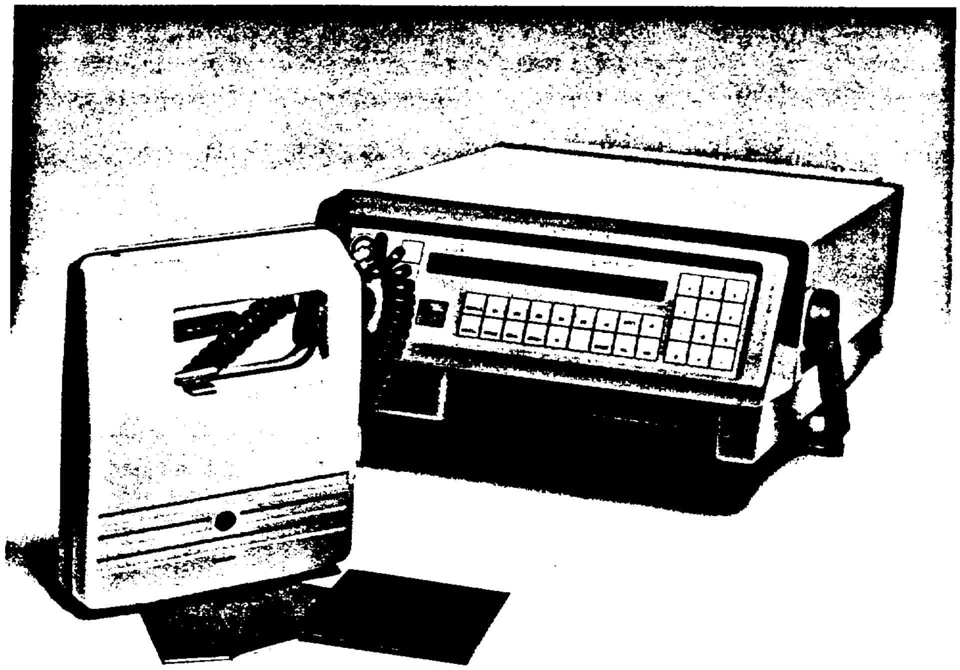
The proposed action is not a major action as defined by 10 CFR Part 51.5(a)(10) and does not require an environmental impact statement.

## 5. CONCLUSION

Regulatory Guide 1.13 should be updated.

# X-MET

## Portable XRF Analyzer



**RAPID IDENTIFICATION AND ANALYSIS ON THE FIELD**

Metorex

# X-MET XRF analyzers

### INTRODUCTION

Fast and accurate material identification is required in many areas of the metals industry, including production, fabrication, inventory control and scrap sorting. X-ray fluorescence spectrometry has, over the last twenty years, gained the recognition of metallurgists as a significant tool in material identification. The speed, reliability, and non-destructiveness of x-ray fluorescence spectrometry make it suitable not only for laboratory applications, but also for field and plant use.

The successful expansion of x-ray ...scence analysis from laboratory ...at environments was prompted by the development of portable analyzers made possible by:

(i.) the use of small, sealed radio-isotope sources used to excite the characteristic x-rays of the sample;
(ii.) the availability of powerful microprocessors; and
(iii.) the use of rechargeable batteries to make the instrument independent of AC power.

Metorex's X-MET x-ray fluorescence analyzer makes use of these developments as well as the most recent advances in microprocessor technolo... The X-MET makes it possible to ...m complex and simultaneous a...ysis of the x-ray spectra from the sample using only a battery powered x-ray analyzer. The X-MET also allows for data storage and processing, a task previously assigned to an off-line computer.

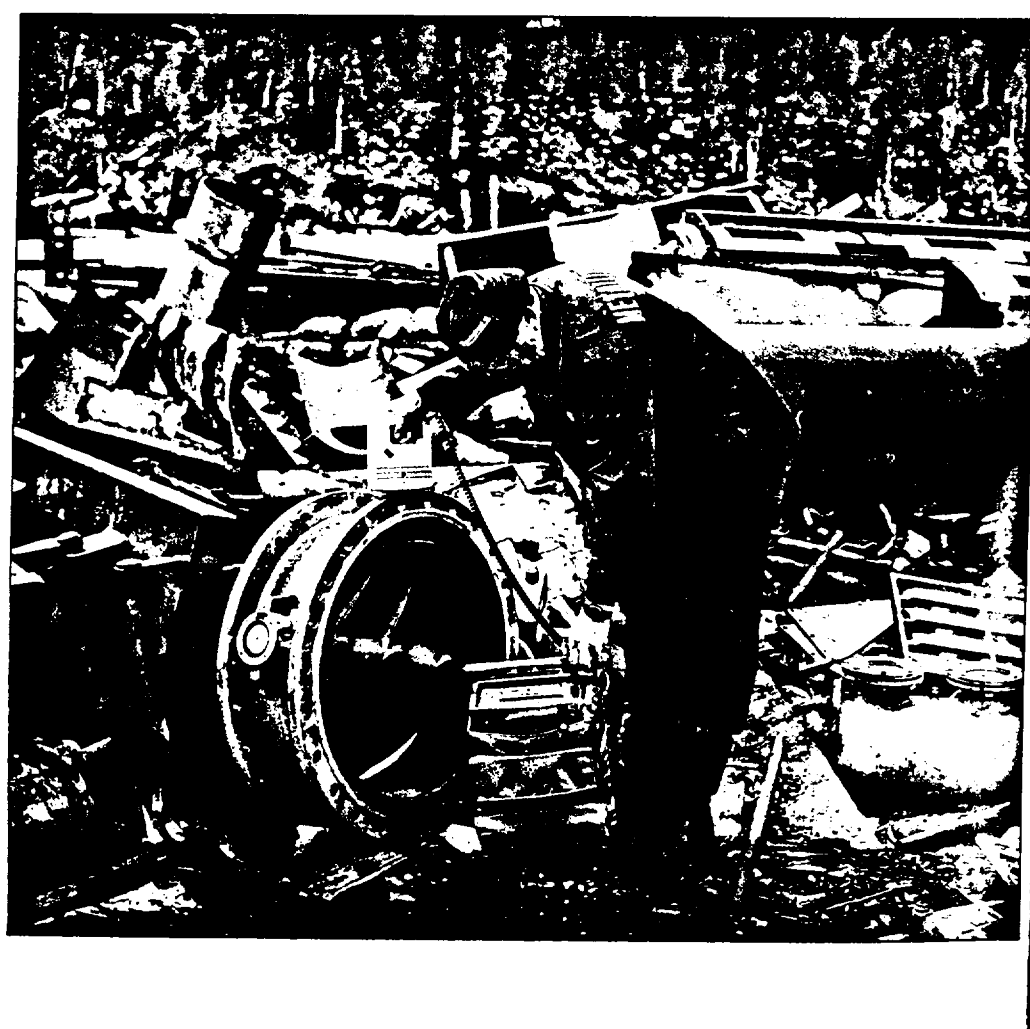
A careful examination of the specifications of thousands of alloys currently used reveals that there are 40 to 50 elements involved in the alloying process, with 10 to 20 typically present in any given alloy. In comparison, the X-MET is capable of measuring all 80 of the elements in the periodic table from atomic number 13, aluminum, through atomic number 92, uranium.

Nbr 1 thru 12
H, He, Li, Be, B, C, N, O, F, Ne, Na, Mg.

Unlike other systems which are limited to 21 or fewer elements and fewer types of alloys, the X-MET is extremely versatile. This advanced design truly represents a significant benefit of the new generation portable alloy analysis systems.

The state-of-the-art X-MET portable alloy analysis system is powerful yet simple to use. It can be factory calibrated to provide a direct readout of alloy name in five seconds or less. Many of the 30,000 alloy types in use today are applicable, including:

- ☐ Stainless and High Temperature Steels
- ☐ Chrome Moly Steels
- ☐ Tool Steels
- ☐ Alloy Steels (with greater than 1 % of either Cr, Ni, Cu, Mn, Mo)
- ☐ Nickel Based Alloys (Inconels, Hastelloys, Monels, etc.)
- ☐ Cobalt Based Alloys (Stellites, Haynes, etc.)
- ☐ Copper Based Alloys (Brasses, Bronzes, Cupro-nickels, etc.)
- ☐ Titanium Based Alloys
- ☐ Aluminum Based Alloys
- ☐ Magnesium Based Alloys
- ☐ Zinc and Lead/Zinc Alloys
- ☐ Exotics (Zirconium Alloys, Molybdenum Alloys, etc.)



## ALLOY IDENTIFICATION

Alloy identification can be defined as a process of ascertaining those characteristics of a given material by which it is definitively recognizable or known.

Various techniques have been used over the years for alloy sorting or identification. The traditional ones include color recognition, magnetism, spark testing, differences in apparent density, and chemical spot tests. More sophisticated methods are based on thermoelectricity and optical emission spectroscopy. In general, these methods require an experienced operator to complete the identification, based on the results of the measurement. The same applies to a conventional full scale chemical analysis of an alloy, which must be followed by a search through composition tables to find the matching alloy name or grade designation.

The portable, microprocessor-based X-MET offers a real breakthrough by relieving the operator from decision making. All that is necessary for analysis is to expose the sample to the instrument for a few seconds, and then read the final identification from the display or printout. Search-match technology is employed which eliminates the need for analysis and judgement procedures.

The X-MET provides direct storage of up to 400 precalibrated alloy signatures, and easy replacement by the user can be done on the spot as new alloy identification needs arise. Reference signatures may be custom named for maximum user convenience. Labels such as bin number, serial number, melt number, etc. may be used in place of, or in addition to, alloy common name and/or alloy proprietary name.

In addition to a rapid and positive identification, the X-MET is capable of providing alloy elemental composition [i.e., concentration of major alloying elements displayed as percentages with their respective element symbol(s)] in about 20 seconds, depending on precision and accuracy needed.

With mislabeling of delivered alloy materials occurring from a few percent up to 10 % or more, and with the chance of mixups on production floors, in salvage operations, on jobsites, etc., the economic and product liability concerns increasingly justify investment in such rapid, positive identification devices.

## PORTABLE X-RAY FLUORESCENCE ANALYSIS

X-ray fluorescence spectrometry is a comparative analytical technique which utilizes the physical principles of the interaction of x-rays or gamma rays with matter. When a sample is exposed to a beam of low energy (1 to about 100 keV) x-rays or gamma rays, the main result is excitation in the sample of the characteristic x-rays of its elements. It is therefore possible to analyze the sample both *qualitatively* (recognition of elements by their unique x-ray patterns) and *quantitatively* (amount of element in the sample is proportional to the intensity of its characteristic x-rays).

The X-MET portable x-ray analyzer, configured for alloy analysis and identification, consists of a hand-held probe and an electronic unit. The probe contains an x-ray source to excite the sample, and a detector which resolves the x-rays and measures their intensities. The electronic unit accepts the signal from the probe, performs all necessary data processing and displays the result. It also contains the power supplies, operator interface and an I/O port for peripherals such as a printer, data logger or personal computer (PC).

The preferred source for a hand-portable instrument is a sealed radioisotope which emits x-rays or low energy gamma rays. Such sources are rugged, free from drift problems and very compact. Typical sources are only 8 mm in diameter by 5 mm thick, weighing about 2 grams. Their output is about 6 orders of magnitude less than that of an x-ray tube, which results in only minimal potential radiation hazard (for the same reason, the use of classic wavelength-dispersive crystal spectrometers with high x-ray tube intensities and high power requirements is impractical).

For optimum performance, x-rays must be measured with high geometrical efficiency, and the detector must be capable of discriminating between x-rays from neighboring elements without further significant loss of x-ray photons. Gas-filled proportional counters have proven themselves over the years as the most reliable detectors used in portable x-ray analyzers.

Until recently, the resolution of proportional counters has not been good enough to avoid the need for balanced filters. However, new developments in proportional counter technology (as used in the X-MET) have yielded detectors which significantly improve the resolution (12-14 % for the MnK line). This, coupled with a superior microprocessor (Motorola 68000) for spectral processing, has resulted in the availability of the hand-portable X-MET x-ray fluorescence analyzer capable of simultaneous multielement analysis.

# X-MET 880

## THE X-MET SYSTEM

With the X-MET portable alloy analysis system, measurements are totally non-destructive and can be made under extremes of environmental conditions ranging from high dust indoor environments to very cold, hot or wet outdoor environments.

The unit features a slim-line lightweight weatherproofed probe and hermetically sealed electronic unit totalling only 8.5 kg. It is designed to fit in a small water repellant backpack for user convenience in field transport. Power for field operations is received [illegible] long lasting lead gel-cell, plug-in [illegible]geable batteries that easily last [illegible] 10 hours of continuous use, without the disadvantage of "memory" effect such as is typical of Ni-Cad type battery systems.

The X-MET system has proven its worth in hundreds of installations throughout the world. It has prevented thousands of costly mix-ups and potential liability problems by providing decision-making data for metallurgical specification analysis.

## KEY FEATURES

[illegible]eys to such a wide range of [illegible]ilities and such high performance in a portable system are:

1.) The high resolution proportional detector which gives good performance and high reliability at ambient temperatures without the need for x-ray filters or liquid nitrogen dewars.
2.) The microprocessor/software/electronics package which takes full advantage of the fundamental improvements in detection capability, while allowing simple straightforward man/machine interfacing and providing powerful search/match pattern recognition techniques for alloy identification. In addition, the microprocessor provides the capability for on-line computation of elemental alloy composition.

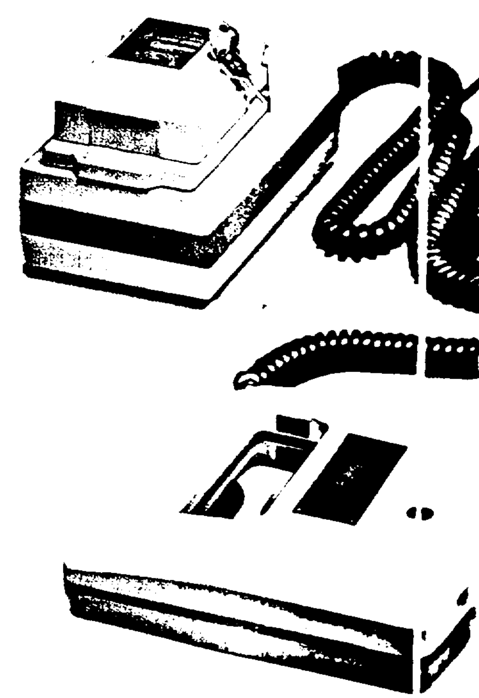
## FIELD PROGRAMMING

With a fully calibrated unit (as supplied from the factory) operator training is minimal, and routine use is extremely simple. However, unlike other portable alloy analyzers of earlier design, the X-MET allows for easy and rapid field reprogramming. A special code provides operator access to programming functions which allow on-the-job customized calibrations for either the identification mode (alloy type) or the assay mode (composition readout). Addition of references in the I.D. mode is extremely simple. (Type in "ADD", then measure a reference standard for 100 to 200 seconds). Calibration training, to allow the user to create special or custom calibrations in the assay mode, is provided by the X-MET representatives worldwide.

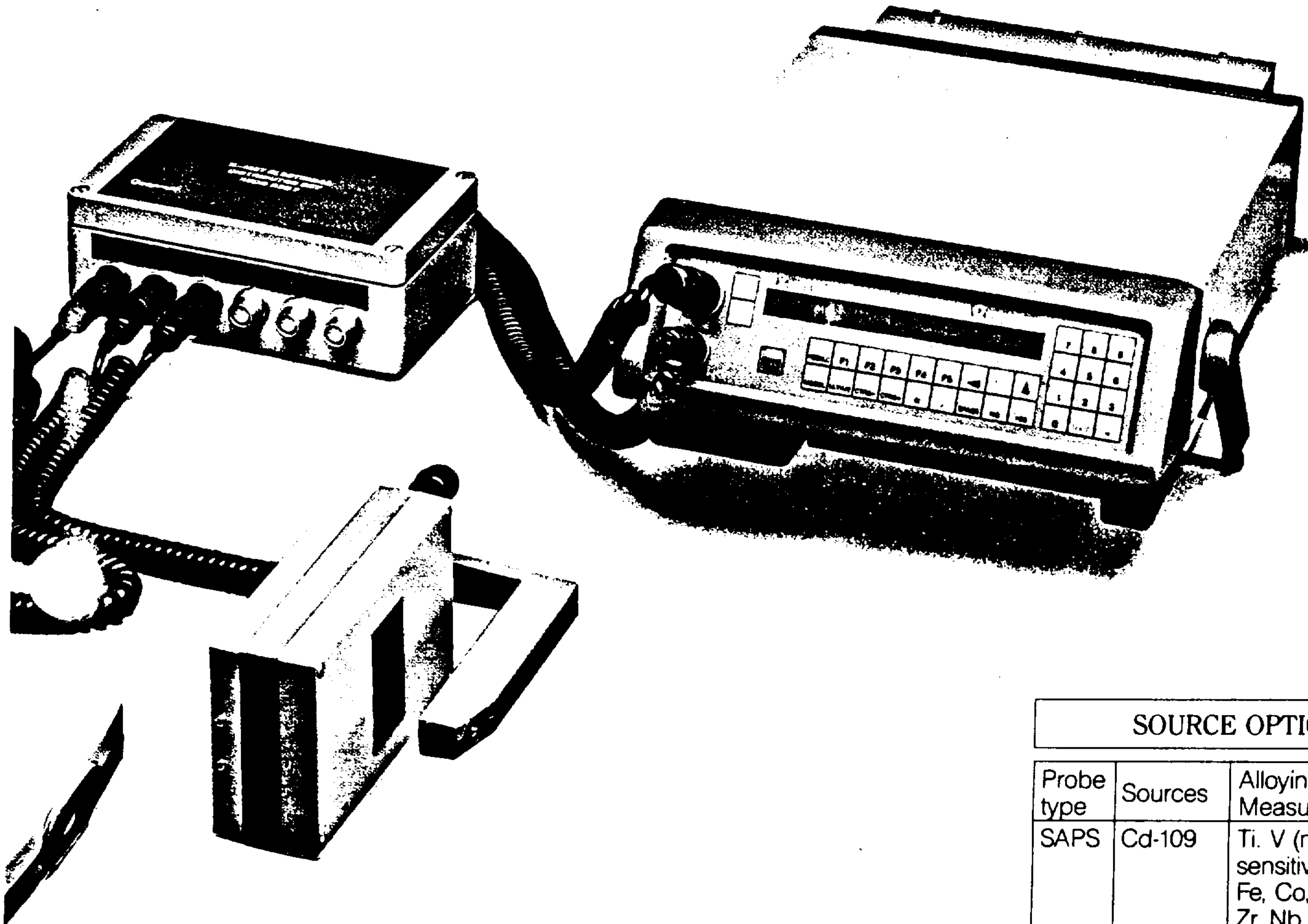
Training required

## OPTIONAL ITEMS

To facilitate field programming, Metorex offers a set of 100 reference alloys, all of which include certificates of analysis showing the participating labs and verifying each laboratory's method based on NIST (National Institute of Standards and Technology) traceability. The set covers the 100 most commonly used alloys, that also represent the optimum calibration suite for each alloy type. (Request Brochure No. BNRM-1 for further alloy standards information.)



To dokument the data from X-MET. Metorex offers a lightweight, portable, battery operated terminal/printer, which plugs directly into the X-MET. For further data handling needs, the X-MET results may be collected on a plug-in data logger, for later readout, or may be interfaced directly to a PC system. Metorex can supply standard software to facilitate use and enhance PC data capture, as well as, data reduction.



| SOURCE OPTIONS | | |
|---|---|---|
| Probe type | Sources | Alloying Elements Measured |
| SAPS | Cd-109 | Ti. V (moderate sensitivity), Cr, Mn, Fe, Co, Ni, Cu, Zn, Zr, Nb, Mo, Hf, Ta, W, Pb |
| DOPS | Cd-109, Fe-55 | Same as a single Cd-109 source, plus excellent performance for Ti, V |
| DOPS | Cd-109, Am-241 | Same as a single Cd-109 source, plus Rh, Pd, Ag, In, Cd, Sn, Sb |
| DOPS | Cm-244, Am-241 | Same as Cd-109/Am-241 dual source w/slightly reduced sensitivity for Nb and Mo, increased sensitivity for Ti, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Zr, Hf, Ta, W, Pb |
| SLPS | Fe-55 | Al, Si, P, S, Ti*, V*, Cr* |

SAPS = SURFACE ANALYSIS PROBE
DOPS = DOUBLE SOURCE PROBE
SLPS = SURFACE LIGHT ELEMENT PROBE
*SLPS with Fe-55 provides the highest possible sensitivity (see table 6).

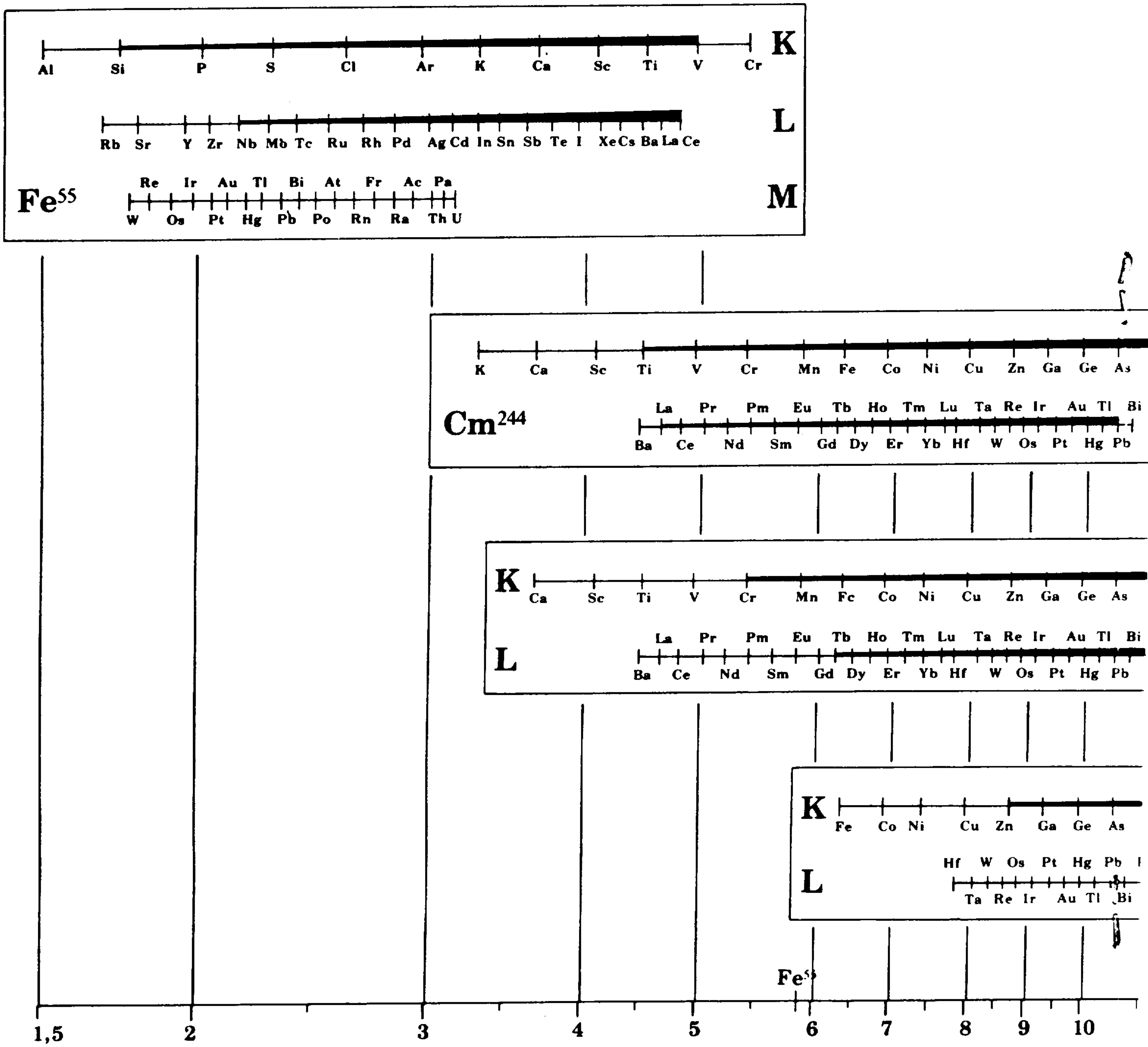
## EXCITATION SOURCE OPTIONS

The system is routinely configured with a Cd-109 excitation source which provides excellent performance for common alloying elements such as chromium (Cr), manganese (Mn), iron (Fe), cobalt (Co), nickel (Ni), copper (Cu), zinc (Zn), niobium (Nb), molybdenum (Mo), zirconium (Zr), tungsten (W) and lead (Pb). A dual-source probe is also available which can contain two excitation sources, such as Cd-109 and Fe-55. With this combination, the performance of the Cd-109 excitation source, for the elements listed above, is maintained; while the elements titanium (Ti) and vanadium (V), which are more efficiently measured using the Fe-55 excitation source, are brought into the "excellent performance" category.

Another example of the benefit of a dual-source probe is the extended element analysis range offered by the combination of Am-241 and Cd-109. This source combination extends the elemental analysis range so that it includes tin (Sn) in, for example, copper and titanium based alloys. The combination of Am-241 and Cm-244 sources allows analysis for elements in the same spread as the Am-241/ Cd-109 range, but with slightly reduced sensitivity for Nb and Mo and somewhat increased sensitivity for Ti, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, W and Pb. In addition to the important capability of Sn analysis, combining the Am-241 source with either Cd-109 or Cm-244 allows analysis for other so-called "heavier"** elements, which may be important for some alloys.

** K shell x-ray energies above 18 keV are considered to arise from "heavy" elements. For example, atomic numbers 44 thru 56 (Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Te, I, Xe, Cs and Ba).

# Source Selection for Different Elements

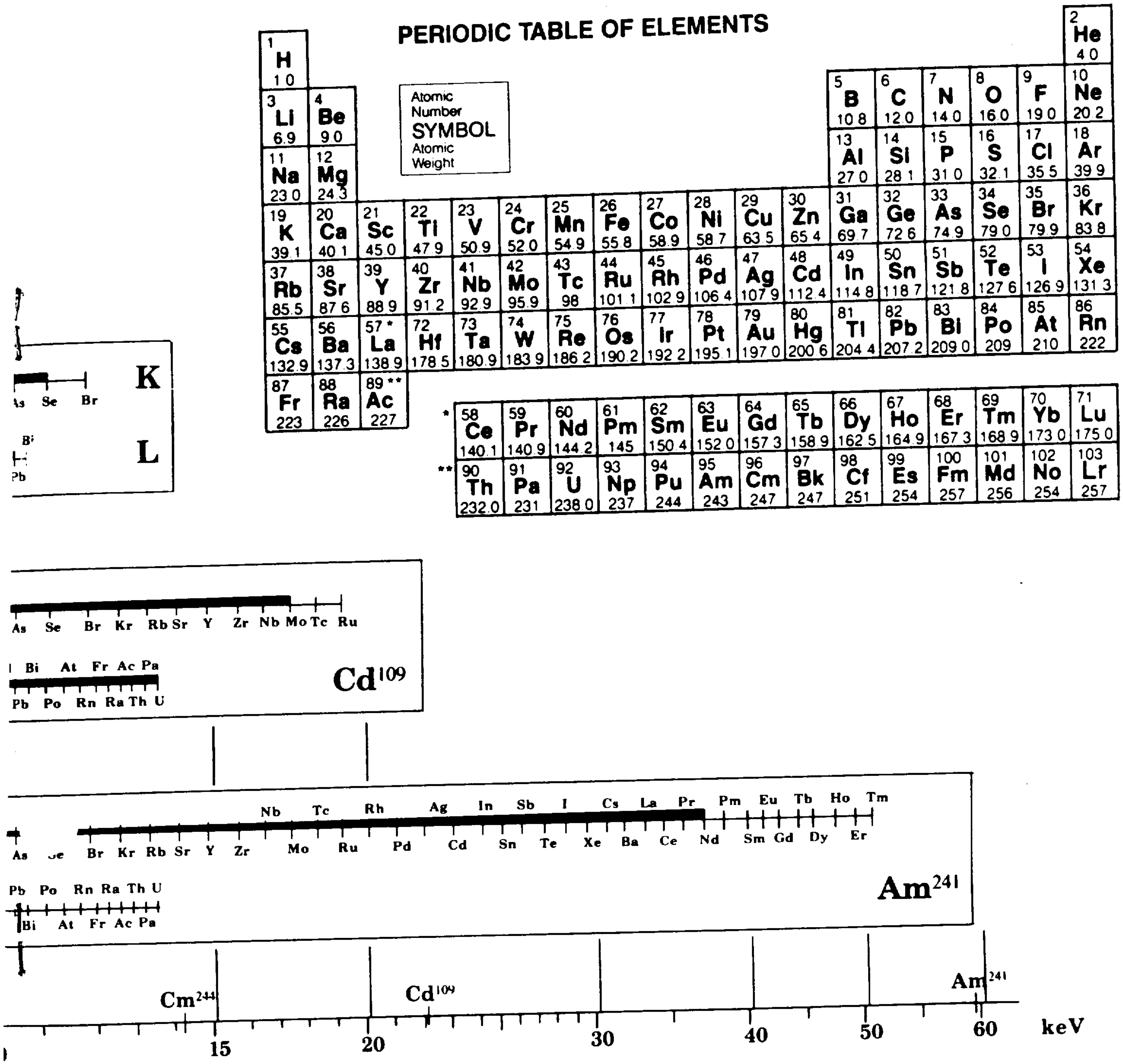



## CHOICE OF SOURCE

To span the whole element range three different isotope sources are needed. Four sources with different activities are available.

The elements shown on the emphasized line comprise the elements for which the source is best suited. The elements on the unemphasized line can be analyzed, though normally with reduced accuracy.

The basic rule for selecting the source with the help of the table is that the K-lines in the emphasized areas are to be preferred.




## SOURCE SPECIFICATION

| Isotope | Half-life | Emission | Preferred Element Ranges K-lines | Preferred Element Ranges L-lines | Detector |
|---|---|---|---|---|---|
| Fe-55 | 2.7 years | Mn K X-rays | Si-V | Nb-Ce | Ne |
| Cm-244 | 17.8 years | Pu L X-rays | Ti-Se | La-Pb | Ar |
| Cd-109 | 1.3 years | Ag K X-rays | Cr-Mo | Tb-U | Ar |
| Am-241 | 433 years | Gamma rays at 59.6 keV | Zn-Nd | Hf-U | Ar |

Source activities are in the range 50—4000 MBq with photon outputs of $10^6$—$5 \times 10^7$ photons/s/sr.

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## APPLICATIONS

Examples of Applications are:

- ☐ Incoming Inspection
- ☐ On-Site Alloy Verification
- ☐ Quality Control
- ☐ Stock Control
- ☐ Scrap Upgrade and Classification
- ☐ Melt Analysis
- ☐ Weld Analysis
- ☐ Maintenance Assessment
- ☐ Construction Site PMI (Positive Material Identification)

Products analyzed include all sizes, shapes, and finishes.

Examples of Products Analyzed are:

| | |
|---|---|
| Sheets | ☐ Blades |
| Ingots | ☐ Fasteners |
| Billets | ☐ Valves |
| ☐ Castings | ☐ Tanks |
| ☐ Plates | ☐ Sludges |
| ☐ Rods | ☐ Turnings |
| ☐ Tubes | ☐ Powders |
| ☐ Bars | ☐ Liquid Digestions |
| ☐ Bolts | ☐ Cutting Oils |

For most products, little or no sample preparation is required: simply place the probe on the material, pull the trigger for a few seconds, then read out the results.

Examples of Applicable Industries include:

- Fossil and Nuclear Power
- Metallurgical Manufacturing Industry
- ☐ Metal Service and Distribution Centers
- ☐ Metal Scrap Recycling Operations
- ☐ Foundries
- ☐ Analysis Service Labs
- ☐ Chemical Process Industries
- ☐ Construction Engineering
- ☐ Refining and Petrochemical
- ☐ Pulp and Paper
- ☐ Metals Fabrication
- ☐ Military Hardware

## UNIQUE CAPABILITIES

The standard alloy system consists of a 1.5 kg trigger-actuated probe with 6×21 mm measurement area (other sizes optional), an excitation source (or sources with dual source probe) and a 7 kg microprocessor-based electronic unit which processes and stores all data. The probe can measure sizes both smaller and larger than the measuring aperture while maintaining correct identification (I.D.) results.

Although the probe must be in contact with the sample and must cover the measuring aperture for quantitative results (elemental composition), the alloy sort (I.D.) mode results are automatically corrected for undersize samples (below 6×21 mm) and non-contacted samples (up to 19 mm away from the probe). One extreme example of an undersize sample that identifies correctly in 5 seconds with no special procedure (such as taping a bundle of rods together) is the separation of 0.8 mm diameter Inconel welding rods.

The standard alloy system can be upgraded any time by the addition of a Surface Light Element Probe. The unique capability to analyze "light"*** elements with a portable analyzer with no "special" consideration (such as helium purge or vacuum path) allows such previously unattainable field analyses as sulphur (S) in steel, silicon (Si) in aluminum, Si and Ti in nickel based alloys, etc.

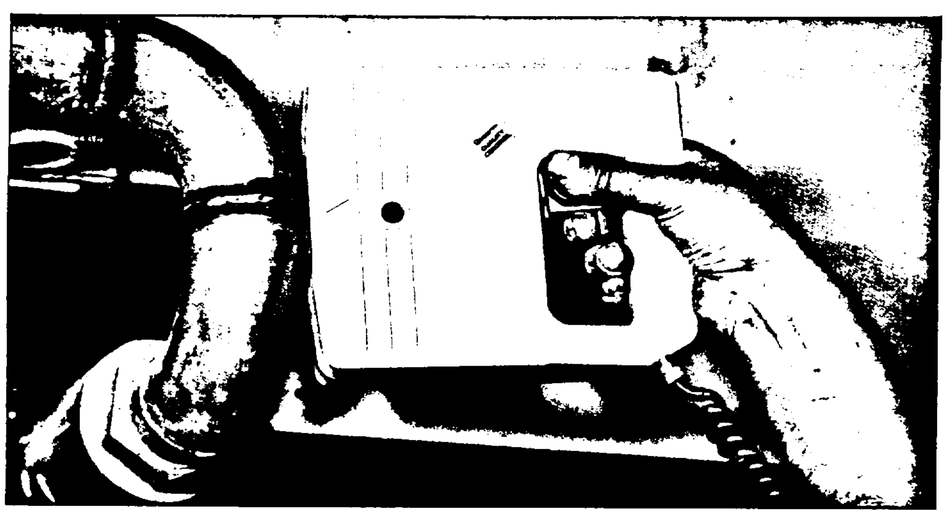
It is well known that the stainless steels 303 and 304 differ only by 0.3 % sulphur, which makes separation of these two grades extremely difficult and challenging. However, the identification mode handles this task very well with the use of a surface light element probe and a Fe-55 source. With this configuration, the X-MET can positively identify 303 from 304 with 100 % success. A similar example is the pair 410 and 416 stainless steel, which also differs by 0.3 % sulphur and can be handled in the very same manner. It should be noted that sulphur and the other elements of atomic number greater than 12 can be measured with this probe using an air path.

In one case, this light element probe proved its excellent sensitivity for titanium in steels by analyzing the residual metal particles in sandpaper and consistently identifying 304-vs-321 correctly in 5 seconds based on the Ti signal from the alloy particles retained on the sandpaper (request the detailed report from your sales agent). With such high sensitivity for titanium, the task of separating 304 from 321 can be accomplished in just one (1) second with 100 % confidence.

Using a standard surface analysis probe with Cd-109 source, the sorting routine has proved itself also to work with a 100 % success rate on separation of stainless steel 303 and 303Se, where the only difference is 0.3 % selenium.

## PERFORMANCE

Examples of performance for a wide range of applications are given in the following tables. Note that the data given represent typical precision and RMS values. In many cases, even better values can be achieved.



*** K shell x-ray energies below 5 keV are considered to be those of the light elements, for example, atomic numbers 13-23 (Al, Si, P, S, Cl, Ar, K, Ca, Sc, Ti and V).

## TABLE 1. SUMMARY OF THE RESULTS OF IDENTIFICATION PROCEDURE

| Alloy Group In Model | Measured Elements | Identification Results for Typical Useage % Feasible |
|---|---|---|
| Nickel Alloys (21 ref.) | Ti, Cr, Fe, Co, Ni, Cu, Nb, Mo, W | 100.0 |
| Copper Alloys (15 ref.) | Mn, Fe, Ni, Cu, Zn, Pb, Sn | 90 to 100 |
| Stainless and High Temp. Steels (26 ref.) | Ti, Cr, Mn, Fe, Co, Ni, Cu, Nb, Mo | 90 to 100 |
| Cr/Mo Steels (6 ref.) | Cr, Fe, Ni, Mo | 95 to 100 |
| Carbon and Low Alloy Steels (9 ref.) | Cr, Fe, Ni, Mo | 65 |
| Titanium Alloys (16 ref.) | Ti, V, Cr, Mn, Cu, Zr, Mo | 95 to 100 |
| Aluminum Alloys (8 ref.) | Ti, Cr, Mn, Fe, Cu, Zn | 90 to 100 |

Within-the-group identification

Several individual models were set up for various alloy groups using a probe equipped with a 5 mCi Cd-109 source. In each model first a library of reference alloys was created by measuring each reference for 200 sec. Then each reference sample was measured, as an unknown, at least ten times for 5 sec. and the percentage of correct or incorrect identifications was recorded. The percent correct or incorrect identification within each alloy group was calculated for all alloys tested within a given alloy group. The results are listed in the above Table.

## TABLE 2. X-MET SYSTEM PERFORMANCE DATA FOR CARBON, LOW ALLOY AND Cr/Mo STEELS

MEASUREMENT CONDITIONS:
Probe: DOPS; Slot Aperture, Dual Source
Meas. Time: 300 sec.

| Source | Cm-244 | | | | | Cd-109 |
|---|---|---|---|---|---|---|
| Element | Cr | Mn | Fe | Ni | Cu | Mo |
| Concentration Range | 0—9 % | 0—1 % | 90—100 % | 0—3.5 % | 0—1 % | 0—1 % |
| RMS Error a) | .07 % | .13 % | .45 % | .2 % | .06 % | .015 % |
| Precision of Measurement b) | .06 % | .13 % | .24 % | .1 % | .015 % | .010 % |

a) Root Mean Square Error ± around the calibration line fitted (LSQ Method) to the experimental data points.
b) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

### TABLE 3. X-MET SYSTEM PERFORMANCE DATA FOR HIGH TEMPERATURE AND STAINLESS STEELS

MEASUREMENT CONDITIONS:
Probe: DOPS; Slot Aperture, Dual Source
Meas. Time: 100 sec.

| Source | Fe-55 | Cd-109 | | | | | |
|---|---|---|---|---|---|---|---|
| Element | Ti | Cr | Mn | Ni | Cu | Nb | Mo |
| Concentration Range | 0–2 % | 0–25 % | 0–15 % | 0–35 % | 0–3.5 % | 0–1 % | 0–3.5 % |
| RMS Error a) | .09 % | .30 % | .40 % | .35 % | .14 % | .01 % | .02 % |
| Precision of Measurement b) | .05 % | .15 % | .20 % | .35 % | .14 % | .01 % | .01 % |

a) Root Mean Square Error ± around the calibration line fitted (LSQ Method) to the experimental data points.
b) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

### TABLE 4. X-MET SYSTEM PERFORMANCE DATA FOR NICKEL, COBALT ALLOYS

MEASUREMENT CONDITIONS:
Probe: SAPS; Slot Aperture
Source: Cd-109
Meas. Time: 100 sec.

| Element | Cr | Fe | Co | Ni | Cu | W | Nb | Mo |
|---|---|---|---|---|---|---|---|---|
| Concentration Range | 0–30 % | 0–67 % | 0–60 % | 0–100 % | 0–32 % | 0–15 % | 0–5 % | 0–28 % |
| RMS Error a) | .65 % | .315 % | .80 % | 1.0 % | .40 % | .38 % | .04 % | .18 % |
| Precision of Measurement b) | .15 % | .20 % | .40 % | .5 % | .30 % | .11 % | .02 % | .18 % |

a) Root Mean Square Error ± around the calibration line fitted (LSQ Method) to the experimental data points.
b) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

### TABLE 5. X-MET SYSTEM PERFORMANCE DATA FOR COPPER ALLOYS (BRONZES & BRASSES)

MEASUREMENT CONDITIONS:
Probe: DOPS; Slot Aperture, Dual Source
Meas. Time: 100 sec.

| Source | Cm-244 | | | | | Am-241 | |
|---|---|---|---|---|---|---|---|
| Element | Fe | Ni | Cu | Zn | Pb | Cd | Sn |
| Concentration Range | 0–5 % | 0–30 % | 60–100 % | 0–40 % | 0–8 % | 0–0.1 % | 0–1 % |
| RMS Error a) | .15 % | .35 % | .45 % | .31 % | .3 % | .020 % | .005 % |
| Precision of Measurement b) | .1 % | .25 % | .45 % | .25 % | .1 % | .008 % | .005 % |

a) Root Mean Square Error ± around the calibration line fitted (LSQ Method) to the experimental data points.
b) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

## TABLE 6. X-MET SYSTEM PERFORMANCE DATA FOR ALUMINUM ALLOYS

| Probe Source | SLPS (Ne)/Fe-55 | | | DOPS (Hp Ar) dual source/CM-244[1] + Am-241[2] | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| Time | 240 seconds | | | 60 seconds | | | | | | |
| Elements | Si | Ti | Cr | Mn[1] | Fe[1] | Ni[1] | Cu[1] | Zn[1] | Pb[1] | Sn[2] |
| Concentration Range | 0–12 % | 0–.16 % | 0–.25 % | 0–.6 % | 0–1.2 % | 0–.5 % | 0–5 % | 0–3.5 % | 0–.35 % | 0–.25 % |
| RMS Error a) | .22 % | .0015 % | .02 % | .04 % | .04 % | .03 % | .08 % | .08 % | .06 % | .02 % |
| Precision of Measurement b) | .20 % | .001 % | .005 % | .03 % | .01 % | .004 % | .005 % | .005 % | .005 % | .003 % |

a) Root Mean Square Error ± around the calibration line fitted (LSQ Method) to the experimental data points.
b) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

## TABLE 7. X-MET SYSTEM PERFORMANCE DATA FOR CARBON, LOW ALLOY AND Cr/Mo STEELS

| MEASUREMENT CONDITIONS:<br>Probe: SAPS; Slot Aperture<br>Source: Cd-109<br>Meas. Time: 300 sec. | | | | | |
|---|---|---|---|---|---|
| Element | Cr | Mn | Fe | Ni | Mo |
| Concentration Range | 0—9 % | 0—1 % | 90—100 % | 0—3.5 % | 0—.1 % |
| RMS Error a) | .2 % | .35 % | .45 % | .15 % | .015 % |
| Precision of Measurement b) | .2 % | .2 % | .3 % | .1 % | .010 % |

a) Root Mean Square Error ± around the calibration line fitted (LSQ Method) to the experimental data points.
b) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

## TABLE 8. X-MET SYSTEM PERFORMANCE DATA FOR TITANIUM ALLOYS

| MEASUREMENT CONDITIONS:<br>Probe: SAPS<br>Source: Cd-109<br>Meas. Time: 100 sec. | | | | | | |
|---|---|---|---|---|---|---|
| Element | V | Cr | Ni | Zr | Nb | Mo |
| Concentration Range | 0—8 % | 0—6 % | 0—1 % | 0—5 % | 0—2 % | 0—15 % |
| Precision of Measurement a) | .2 % | .2 % | .3 % | .02 % | .001 % | .03 % |

a) One standard deviation due to counting statistics; value reported is valid for the measurement time given in top of the table.

# Metorex

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PAINOKURKI OY 9/95 Ref. 5100 188-4HE

# Request for Additional Information

**Shearon Harris Power Plant, Docket No. 50-400**

**Request for License Amendment, "Spent Fuel Storage," dated December 23, 1998**

The following request for additional information (RAI) is for the purpose of developing an inspection plan and performing safety evaluations for the spent fuel pool (SFP) 'C' and 'D' piping, as described in the licensee's submittal, by letter dated December 23, 1998.

The term "original construction", as used herein, applies to the construction performed under the licensee's N certificate. The term "weld" applies to welders, weld joint, and all material associated with the weld.

## I. Existing Piping System

A. Detailed description of the proposed change:

1. Provide isometric drawings (isometrics) showing all piping and piping systems within the scope of the proposed alternatives; i.e., for fuel pool cooling and cleanup system (FPCCS) and component cooling water system (CCWS) piping and for continuance of design and construction without an N stamp.

2. Provide weld matrixes that list all the welds (each weld should be uniquely identified and traceable to I.A.1. above) within the scope of the alternatives.

3. In the matrixes, or on the isometrics, identify the piping material (ASME/ASTM Specification), weld material (ASME/ASTM Specification), the existence of all required material documentation, and any specific missing documentation. Identify each missing document for each weld. Identify the method(s) used for reconciliation of each type of missing document (e.g., missing Certified Material Test Report reconstructed with complete chemical analysis run on shavings taken from the material). For the sampling and testing methods used for reconciliation, identify references used for guidance (i.e., NRC DG-1070, ASME, or EPRI). Explain any differences between the sampling/ testing methods and the selected referenced guidance.

4. In the matrixes or on the isometrics, identify inaccessible non-embedded welds and embedded welds (all other welds should be accessible).

5. On the isometrics, indicate the specific location of each weld listed in I.A.2. and identify the boundaries of the systems that are considered safety related. Identify all non safety related items that appear on the isometrics.

6. Identify in the matrixes, or on the isometrics, the welds that will be or have been inspected or reinspected that have Code documentation, welds that have been



OPTIONAL FORM 99 (7-90) 2/24/99

| FAX TRANSMITTAL | # of pages ▶ 5 |
|---|---|
| To Kevin Shaw | From Rich Laufer |
| Dept./Agency Harris Plant | Phone # 301 415-1373 |
| Fax # 919-362-2701 | Fax # 301 415-3061 |

NSN 7540-01-317-7368 5099-101 GENERAL SERVICES ADMINISTRATION inspected that do not have Code documentation, and welds that will be or have been inspected or reinspected not to Code. For the welds that will be or have been inspected or reinspected but not to Code, describe the inspection technique, acceptance criteria, and documentation. Identify the edition and addenda of ASME Code that will be or has been used for the above inspections and reinspections.

7. Identify any non safety related items installed during the original construction that will be upgraded to safety related status by this amendment; e.g., will any of the non safety related ANSI B31.1 piping (Enclosure 8, page 7 of the submittal) be upgraded?

8. Identify any commercial grade items installed during original construction. If dedication was used during original construction, is documentation of the dedication program available for review? Are the dedication packages for items available for review?

9. Identify any commercial grade items requiring dedication that will be used to complete construction.

10. Was the piping system constructed in accordance with a 10 CFR Part 50, Appendix B program? Is the construction Appendix B program documentation available for review? If construction was performed under a different program, identify the program. Is the program documentation available for review?

11. Are the work control procedures and hold point sign-off documents from the original construction available for review? If these documents are required by Code, which documents are missing?

12. Provide a list of the qualified welders who worked on the original construction and identify the ones qualified to weld stainless steel piping. Are historical qualification records for these welders available for review? If not, provide an explanation to support acceptability. For welds missing welder identification, how will weld integrity be established?

B. <u>Applicable regulations for welds and piping systems within the scope of the proposed alternatives</u>

1. Identify the edition and addenda of Code and any code cases that were used for original construction of the welds and piping systems.

# Request for Additional Information

**Shearon Harris Power Plant, Docket No. 50-400**
**Request for License Amendment, "Spent Fuel Storage," dated December 23, 1998**

2. Identify the edition and addenda of Code and code cases that will be used to complete construction of the piping systems. Identify any exceptions to Code requirements and justifications for these exceptions.

3. Identify the edition and addenda of Code and code cases that were or will be used for repair and replacement of welds and piping?

4. Provide a matrix (See I.A.2.) that identifies the specific paragraphs in Code applicable to each weld. Identify documentation deficiencies for each weld. Identify any exceptions to Code requirements? Provide alternatives and justifications for these exceptions.

5. Identify the ASME requirements, including administrative requirements, that were completed prior to stoppage of the original construction of the piping systems. Is documentation of these completed requirements available for review? What ASME data reports were filed and their filing dates?

6. Identify ASME survey inspections conducted prior to stoppage of the original construction of the piping systems. Are documented results available for review?

7. Identify third party inspections (e.g., Hartford, ANI) conducted prior to stoppage of the original construction of the piping systems? Are these reports available for review?

8. With regard to piping system components/services performed by others, are documented validations of these vendors services available for review?

## II Completion of Piping System (General)

1. Does CP&L have an active construction permit for Shearon Harris?

2. Identify the differences between HNP's proposed construction program to complete the SFP C and D and the original construction program under HNP's N certificate. How will these differences be reconciled?

3. Will data packages be prepared?

4. What third party verification is planned?

# Request for Additional Information

**Shearon Harris Power Plant, Docket No. 50-400**

**Request for License Amendment, "Spent Fuel Storage," dated December 23, 1998**

## III. Specific Comments on Submitted Information (Enclosure 6, December 28 Submittal)

1. What was the basis for selecting the four externally accessible field welds for internal examination (p6/7)? Identify these welds in the matrix provided in response I.A.2 above.

2. With reference to the "substantial portion of the embedded piping and field welds" (p7), identify these welds in the matrix provided in response I.A.2 above.

3. Provide the inspection procedure used for remote inspection of embedded welds.

4. With reference to the remote inspection of the embedded welds, identify the critical characteristics that will be verified and the acceptance criteria to be used.

5. Provide the results of the remote inspection with any identified discrepancies.

6. Provide a completed weld data report, representative of those that were discarded (analogous records exist for the licensed unit). Identify the critical characteristics and explain how, in lieu of records, each will be validated (see I.A.3. and I.A.11. above).

7. With reference to the procurement specification (SS-021, Purchasing Welding Materials for Permanent Plant Construction) (p9), did other specifications for other filler materials exist? What assurances are provided that these other filler materials were not used for the embedded piping?

8. Provide any updates/supplements to the Alternative Plan (p 10) as they become available.

9 With referenced to the "large percentage of embedded field welds" that will be inspected (p 10), identify these welds on the matrix provided (see I.A.4. above). Provide technical justification for not examining the remaining welds.

10. Explain what is meant by the statement that internal examination of the embedded welds provides a measure of quality assurance beyond Code requirements (p11). What additional physical or material attributes will be verified?

11. The submittal refers to opinions by Bechtel and Hartford Steam Boiler concerning the benefits in accordance with an N certificate program (p. 12). Are these opinions documented and available for review?

# Request for Additional Information

**Shearon Harris Power Plant, Docket No. 50-400**
**Request for License Amendment, "Spent Fuel Storage," dated December 23, 1998**

12. Provide a matrix comparing the specific ASME Section III requirements with the Corporate QA Program.

13. Provide a matrix comparing the specific ASME Section III requirements with the corresponding Section XI requirements.

14. Provide documentation of the referenced comparison (p12) of approved ASME Section III Construction QA Program Manual with the effective Corporate 10CFR50, Appendix B QA Program.

15. Provide documentation of the supplemental quality assurance requirements that have been developed (p13) specifically for the purpose of addressing differences between ASME Section III quality assurance requirements and the Corporate 10CFR Appendix B QA Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

| FACSIMILE TRANSMISSION | DATE: 4/21/99 |
|---|---|

TO: KEVIN Shaw

FAX NO: 919-362-2701 TEL NO: 919-362-2830

FROM: RICH LAUFER

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

FAX NO.: (301) 415-2102 TEL NO: 301-415-1373

PAGE 1 OF 2 PAGES

REMARKS:

Kevin -

Here are questions for discussion during a conference call in the next few days.

— RICH

REQUEST FOR ADDITIONAL INFORMATION

SHEARON HARRIS SPENT FUEL STORAGE CAPACITY INCREASE

Reactor Systems Branch

1) Although the burnup criteria for storage in Pools C or D will be implemented by administrative procedures to ensure verified burnup prior to fuel transfer into these pools, an administrative failure should be assumed and evaluation of a fuel assembly misloading event (i.e., a fresh PWR assembly inadvertently placed in a location restricted to a burned assembly as per TS Fig. 5.6.1), should be analyzed.

2) How will the burnup requirements needed to meet TS Fig. 5.6.1 be ascertained for fuel assemblies shipped from other PWR plants (Robinson)?

3) The fuel enrichment tolerance is specified in Section 4.5.2.5 as +0.0/-0.05. Why isn't a positive tolerance of +0.05 assumed (i.e., 5.0+0.05 weight percent U-235)?

4) Justify that the allowance that was assumed for possible differences between the fuel vendor and the Holtec calculations is sufficient to also encompass burnup calculational uncertainties.

5) The summary of criticality safety calculations shown in Tables 4.2.1 and 4.2.2 indicate that the total uncertainty is a statistical combination of the manufacturing tolerances but do not indicate methodology biases and uncertainties. Were these included?

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 24, 1999

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ALTERNATIVE PLAN FOR SPENT FUEL POOL COOLING AND CLEANUP SYSTEM PIPING - SHEARON HARRIS NUCLEAR POWER PLANT (TAC NO. MA4432)

Dear Mr. Scarola:

By letter dated December 23, 1998, you requested a license amendment to revise Shearon Harris Nuclear Power Plant Technical Specification (TS) 5.6, "Fuel Storage," to increase the spent fuel storage capacity by adding rack modules to pools 'C' and 'D.' Enclosure 8 of your submittal provided a detailed description of the proposed alternatives to demonstrate compliance with ASME B&PV Code requirements for the cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i).

During the course of its review, the NRC staff has determined that additional information is necessary to complete its review. The enclosed request for additional information regarding your proposed alternative plan was discussed with your Licensing staff on March 9, 1999. A mutually agreeable target date of April 30, 1999, for your response was established. If circumstances result in the need to revise the target date, please call me at the earliest opportunity.

Sincerely,

Richard J. Laufer

Richard J. Laufer, Project Manager
Project Directorate II-3
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosure: As stated

cc w/encl: See next page

cc:

Mr. William D. Johnson
Vice President and Corporate Secretary
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

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c/o U.S. Nuclear Regulatory Commission
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New Hill, North Carolina 27562-9998

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Assistant Attorney General
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Executive Director
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Chairman of the North Carolina
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Vice President-Harris Plant
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Mr. Vernon Malone, Chairman
Board of County Commissioners
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Board of County Commissioners
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# Request for Additional Information

**Shearon Harris Nuclear Power Plant**
**alternative plan for spent fuel pool cooling and cleanup system piping**

The term "original construction," as used herein, applies to the construction performed under the licensee's N certificate. The term "weld" applies to welders, weld joint, and all material associated with the weld.

## I. Existing Piping System

A. Detailed description of the proposed change:

1. Provide isometric drawings (isometrics) showing all Code-related piping and piping systems within the scope of the proposed alternatives; i.e., for fuel pool cooling and cleanup system (FPCCS) and component cooling water system (CCWS) piping. Provide isometric drawings to be used for continuance of design and construction without an N stamp.

2. Provide weld matrixes that list all the welds (each weld should be uniquely identified and traceable to I.A.1. above) within the scope of the alternatives.

3. In the matrixes, or on the isometrics, identify the piping material (ASME/ASTM Specification), weld material (ASME/ASTM Specification), the existence of all required material documentation, and any specific missing documentation. Identify each missing document for each weld. Identify the method(s) used for reconciliation of each type of missing document (e.g., missing Certified Material Test Report reconstructed with complete chemical analysis run on shavings taken from the material). For the sampling and testing methods used for reconciliation, identify references used for guidance (i.e., NRC DG-1070, ASME, or EPRI). Explain any differences between the sampling/ testing methods and the selected referenced guidance. For chemical analysis, identify sample size and chemical analysis (mean and standard deviation for each element) for each analyzing technique.

4. In the matrixes or on the isometrics, identify inaccessible non-embedded welds and embedded welds (all other welds should be accessible).

5. On the isometrics, indicate the specific location of each weld listed in I.A.2. and identify the boundaries of the systems that are considered safety-related. Identify all non-safety-related items that appear on the isometrics.

6. Identify in the matrixes, or on the isometrics, the welds that will be or have been inspected or reinspected that have Code documentation, welds that have been inspected that do not have Code documentation, and welds that will be or have been inspected or reinspected not to Code. For the welds that will be or have been inspected or reinspected but not to Code, describe the inspection technique, acceptance criteria, and documentation. Identify the edition and addenda of ASME Code that will be or has been used for the above inspections and reinspections.

# Request for Additional Information

**Shearon Harris Nuclear Power Plant**
**alternative plan for spent fuel pool cooling and cleanup system piping**

7. Identify any non-safety-related items installed during the original construction that will be upgraded to safety-related status by this amendment; e.g., will any of the non-safety-related ANSI B31.1 piping (Enclosure 8, page 7 of the submittal) be upgraded?

8. Identify any commercial grade items requiring dedication that were installed during original construction. For these items, is documentation of the dedication program available for review? Are the dedication packages for items available for review?

9. Identify any commercial grade items requiring dedication that will be used to complete construction.

10. Was the piping system constructed in accordance with a 10 CFR Part 50, Appendix B program? Is the construction Appendix B program documentation available for review? If construction was performed under a different program, identify the program. Is the program documentation available for review?

11. Are the work control procedures and hold point sign-off documents from the original construction available for review? If these documents are required by Code, what documents are missing?

12. Provide a list of the weld procedure specifications (WPS) used and their procedure qualification records (PQRs). For welds missing welder identification, how will weld integrity be established?

B. <u>Applicable regulations for welds and piping systems within the scope of the proposed alternatives</u>

1. Identify the edition and addenda of Code and any Code cases that were used for original construction of the welds and piping systems. If not the same for all the welds, identify the Code requirements for each weld or groups of welds.

2. Identify the edition and addenda of Code and Code cases that will be used to complete construction of the piping systems. Identify any exceptions to Code requirements and justifications for these exceptions.

3. Identify the edition and addenda of Code and Code cases that were or will be used for repair and replacement of welds and piping.

4. Provide a matrix (See I.A.2.) that identifies the specific paragraph in Code that is applicable to missing weld documents. Identify documentation deficiencies for each weld. Identify any exceptions to Code requirements. Provide alternatives and justifications for these exceptions.

# Request for Additional Information

**Shearon Harris Nuclear Power Plant**

**alternative plan for spent fuel pool cooling and cleanup system piping**

5. Identify the ASME requirements, including administrative requirements, that were completed prior to stoppage of the original construction of the piping systems. Is documentation of these completed requirements available for review? What ASME data reports were filed and their filing dates?

6. Identify ASME survey inspections conducted prior to stoppage of the original construction of the piping systems. Provide documentation for representative internal/external audits conducted during the peak construction periods for the welds in question (1978-1979), particularly in the areas of work control, welding, material traceability, and records.

7. Identify third party inspections (e.g., Hartford, ANI) conducted prior to stoppage of the original construction of the piping systems. Provide a representative sample of documentation for these inspections.

8. With regard to piping system components/services performed by others, provide documented validations of these vendor services. Provide the documentation of audits of the supplier of prefabricated piping.

## II Completion of Piping System (General)

1. Identify the differences between HNP's proposed construction program to complete the SFP C and D and the original construction program under HNP's N certificate. How will these differences be reconciled?

2. Will data packages be prepared?

3. What third party verification is planned?

## III. Specific Comments on Submitted Information (Enclosure 6, December 28 Submittal)

1. What was the basis for selecting the four externally accessible field welds for internal examination (p6/7)? Identify these welds in the matrix provided in response I.A.2 above.

2. With reference to the "substantial portion of the embedded piping and field welds" (p7), identify these welds in the matrix provided in response I.A.2 above.

3. Provide a summary of the inspection procedure used for remote inspection of embedded welds.

# Request for Additional Information

**Shearon Harris Nuclear Power Plant**

**alternative plan for spent fuel pool cooling and cleanup system piping**

4. With reference to the remote inspection of the embedded welds, identify the critical characteristics that will be verified and the acceptance criteria to be used.

5. Provide the results of the remote inspection with any identified discrepancies.

6. Provide a completed weld data report, representative of those that were discarded (analogous records exist for the licensed unit). Identify the critical characteristics and explain how, in lieu of records, each will be validated (see I.A.3. and I.A.11. above).

7. With reference to the procurement specification (SS-021, Purchasing Welding Materials for Permanent Plant Construction) (p9), did other specifications for other filler materials exist? What assurances are provided that these other filler materials were not used for the embedded piping?

8. Provide any updates/supplements to the Alternative Plan (p 10) as they become available.

9 With referenced to the "large percentage of embedded field welds" that will be inspected (p 10), identify these welds on the matrix provided (see I.A.4. above). Provide technical justification for not examining the remaining welds.

10. Explain what is meant by the statement that internal examination of the embedded welds provides a measure of quality assurance beyond Code requirements (p11). What additional physical or material attributes will be verified?

11. The submittal refers to opinions by Bechtel and Hartford Steam Boiler concerning the benefits in accordance with an N certificate program (p. 12). Please provide documented endorsements.

12. Provide a copy of the site ASME Section III QA program used during original construction.

13. Provide a copy of the Corporate QA program that will be used to complete construction. Provide a list of implementing quality control procedures for welder qualification, weld procedures, inspections, documentation, etc.

14. Provide a copy of the supplemental quality assurance requirements developed to augment the Corporate QA program, which was based on review of the approved Construction QA Program at the time of construction versus the existing Corporate QA Program.

# Request for Additional Information
## Shearon Harris Nuclear Power Plant
## alternative plan for spent fuel pool cooling and cleanup system piping

15. Provide documentation of the referenced comparison (p12) of approved ASME Section III Construction QA Program Manual with the effective Corporate 10CFR50, Appendix B QA Program.

16. Provide documentation of the supplemental quality assurance requirements that have been developed (p13) specifically for the purpose of addressing differences between ASME Section III quality assurance requirements and the Corporate 10CFR Appendix B QA Program.

Richard Laufer - TS_harr18.wpd

Page

OPTIONAL FORM 99 (7-90)

**FAX TRANSMITTAL** # of pages ► 2

| | |
|---|---|
| To KEVIN SHAW | From RICK LAUFER |
| Dept./Agency HARRIS Nuclear Plant | Phone # 301-415-1373 |
| Fax # 919-362-2701 | Fax # 301-415-2102 |

Cover Letter Draft:

We are currently reviewing your request, submitted by letter dated December 23, 1998, for a license amendment to place the Harris Nuclear Plant spent fuel pools "C" and "D" in service.

By letter dated April 30, 1999, you provided a response to our request for additional information required to complete the review of the proposed alternative piping plan. In conjunction with review of this information, a telecon was held on August 19, 1999 to further clarify certain, attached items, for which your staff has agreed to docket its response.

- **Weld Material**

1. In Enclosure 4, "Metallurgy Unit Report for Spent Fuel Pool Weld Metal Composition analysis" of our request for additional information (RAI), explain how the Metorex X-Met 880 Alloy Analyzer discriminates between the different standards that you used in your analysis. What are the chemical element ranges associated with the different standards that you used? What determines a match on a particular standard. What chemical elements are not included in the "Match" determination and how are these elements reconciled?

2. Provide assurance that the ferrite numbers are acceptable for A-No. 8 weld wire (ND-2433) used in welds with missing weld wire documentation.

3. In Enclosure 6, "Lab Test Reports," of your response to our RAI, explain the chemical analysis in the Table associated with PQR 6(c), dated 11/15/84, page 2 of 2, laboratory test No. 9-2-149. What roll(s) are associated with the base material, weld, and standard(s)? What criteria was used to determine acceptability.

4. For the piping and welds examined internally, provide a discussion of the examination results. What inspection criteria is used for evaluating the piping and welds for corrosion and fouling? Describe the corrosion and fouling inspection procedure and inspection personnel qualification process. For the embedded welds not examined internally, describe what is preventing their examination.

5. What are the chemical analysis for steel welds 2-CC-3-FW-207, 2-CC-3-FW-208, and 2-CC-3-FW-209?

6. Provide the paper trail that identifies a specific weld material to a specific weld on the isometric drawings, i.e. show that the weld material being verified with the Metorex X-Met 880 was specified for that location. Identify missing documentation that breaks the paper trail, if any.

7. Discuss ~~the~~ chemical analysis and any other analysis performed on the water in the FPCCS and CCWS of the SFP C and D? Where did the water come from? Discuss any differences between the chemical analysis and any other analysis and the original water source. Provide the staff with representative analysis of the water.

8. In Enclosure 8 "Hydrotest Records for Embedded Spent Fuel Pool Cooling Piping and Field Welds," of your response to our RAI, you provided signed hydrostatic test reports for 13

embedded welds. Starting with the signed hydrostatic test report, back track through procedures and program requirements to the point where the missing document(s) were verified as being complete. In another words, identify the specific procedural and program controls requiring verification of completion of the missing documentation (manufacturing/fabrication records, weld data records, updated isometric drawings, and inspections) starting backward from the hydrostatic test report.

9. Identify the concrete pouring procedure that requires checking for the welder symbol and a successful hydrostatic test before pouring.

10. Describe how the liner leak tests support weld integrity for welds 2-SF-8-FW-65 and2-SF-8-FW-66 (Enclosure 3 of your response to NRC's RAI). For these two welds, back track through procedures and program requirements to the point where the missing documents were verified as being completed.

- **Condition of Equipment**

11. What was the condition of layup for the partially completed piping.

12. Describe precautions that were taken to protect system components (e.g., pumps, valves, heat exchangers, piping) from deleterious environmental effects during layup.

13. Summarize the activities being taken to ensure the acceptable quality of equipment before being returned to service.

- **Embedded Welds**

14. Only 6 of the 15 embedded welds will be inspected. Discuss the physical limitations of the inspection equipment that limits inspectability.

15. Why was visual inspection rather than, say, ultrasonic inspection chosen to examine the integrity of the embedded welds?

16. Discuss why the decision not to inspect all of the embedded welds will result in an acceptable level of quality and safety.

- **Post-Modification Testing**

17. Describe the post modification testing to be performed to ensure that the system(s) will satisfy all design requirements. Include description of hydrotests to verify the integrity of the system pressure boundaries, flushing to ensure unobstructed flow

through system components, and preoperational functional testing under design flow/heat loads.

October 5, 1999

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'99 OCT -6

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | |
| | ) | |
| CAROLINA POWER & LIGHT COMPANY | ) | Docket No. 50-400-LA |
| | ) | ASLBP No. 99-762-02-LA |
| (Shearon Harris Nuclear Power Plant) | ) | |
| | ) | |

## CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S RESPONSE TO ORANGE COUNTY'S FIRST SET OF DISCOVERY REQUESTS TO NRC STAFF" in the above captioned proceeding have been served on the following through deposit in the Nuclear Regulatory Commission's internal mail system or as indicated by an asterisk, by first-class mail and by electronic mail (e-Mail) transmission where indicated this 5th day of October, 1999:

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