JNMM

Journal of Nuclear Materials Management

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Peter Dyck

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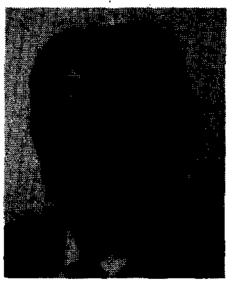
Good Things Are Happening in INMM



In earlier messages, I expressed my optimism about the future of INMM and especially 2001. Many good things are already happening.

I am pleased to see more information exchanged about technical division and chapter activities and among members. Much of this is due to e-mail and your willingness to share information. As a result, we are going to have a stronger organization.

We have already sponsored one very successful division workshop—the Spent Fuel Management Seminar—and have others in the planning stages. Plans for PATRAM (Packaging and Transportation of Radioactive Materials) are moving along very well. The U.S. Department of Energy sponsors this meeting which will be held at the Chicago Hilton and Towers, September 3-7, in cooperation with the International Atomic Energy Agency. This year INMM is hosting PATRAM. Attendance is expected to equal the INMM Annual Meeting attendance. Billy Cole, chair of the INMM Packaging and Transportation Division, deserves credit for INMM being asked to host this meeting. Billy, with the assistance of the INMM headquarters staff, is doing most of the work in planning this meeting. For more information, see the PATRAM Web site at www.patram.org.

Planning for the 42nd INMM Annual Meeting, at the Renaissance Esmeralda Resort, Indian Wells, California, July 15-19, is also moving along well. A wide variety of papers have been chosen for presentation. These, coupled with our strong opening and closing plenary sessions, should make this annual meeting even better than past meetings.

The International Safeguards Division is planning to have a special summer 2001 issue of *JNMM.* They anticipate several papers on integrated safeguards and an associated session at the annual meeting.

I mentioned in my last column that I attended the INMM/ESARDA Workshop on Science and Modern Technology for Safeguards, November 12-16, in Tokyo. The purpose of this workshop was to discuss nuclear safeguards and nonproliferation issues facing the world today. It brought technologists from numerous fields together with safeguards experts for technical communications and technological integration. For more information, see Cecil Sonnier and Steve Dupree's summary on page 4.

A major revelation for me was learning that the need for electrical energy that is acute and rising among East Asian countries. This need will mostly be filled with nuclear-power generated electricity. Large-scale power reactors in Japan and South Korea deliver 30 percent and 43 percent of their nations' electricity respectively. Japan has fifty-two nuclear power reactors and Korea has sixteen in operation and these numbers are increasing.

I also attended the INMM Spent Fuel Management Seminar XVIII, January 10-12, in Washington, D.C. This seminar affords attendees the opportunity to informally and actively participate in the discussions and interface with experts in the research and development, design, engineering, siting, licensing, construction, and operation of spent fuel management equipment and facilities—as well as with program managers and policymakers. It was rewarding to learn more about this important aspect of nuclear materials management. Major objections to the use of nuclear energy in the United States (in addition to safety and the general fear of radiation) were the cost of nuclear-generated electricity and what to do with the nuclear waste. Steve Kraft of the Nuclear Energy Institute reported that the generation of electrical power with nuclear energy was $0.0183 per kilowatt-hour and was now cheaper than coal-, oil-, and gas-generated electrical power. He also pointed out that although we do not have the issue of very long-term storage of nuclear waste material solved, we certainly do know how to transport it and to store it safely in interim-storage sites. The expectation of most of the attendees seemed to be that the long-dormant interest in building new nuclear power plants is about to end. It may even be accelerated in the United States with the current electrical energy crisis in California. This renewed interest in nuclear power will allow INMM safeguards experts to continue to contribute to our nation's well being. The technical papers in this issue of *JNMM* were all presented at the Spent Fuel Management Seminar.

I encourage you to start making plans now to attend the 42nd INMM Annual Meeting. Division meetings and a golf tournament are set for Sunday, July 15. The technical division meetings promise to be more useful than ever. You'll receive more information on the technical division meetings in advance. The hotel registration deadline is June 15 and the reduced preregistration rate for the meeting is available through June 25. Registration and other information about INMM are available on the INMM Web site at http://www.inmm.org.

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Perspectives on Spent-Fuel Management

This edition of the *Journal* has interesting articles addressing waste management issues. They are fun reading. These papers were presented at the INMM Spent Fuel Management Seminar XVIII held in Washington, D.C. in January.

Ed Johnson, chair of the INMM Waste Management Technical Division, suggested these papers for publication in the *Journal.*

The first paper, R&D Program for Interim Storage of Spent Fuel in CRIEPI, by A. Okumura, K. Shirai, and T. Saegusa of the Japan's Central Research Institute of Electric Power Industry, describes some of the efforts underway in Japan. In the second paper, International Atomic Energy Agency Safeguards Impacts on Geological Repositories, Bruce Moran of the U.S. Nuclear Regulatory Commission provides a nice summary of the approaches being considered by the International Atomic Energy Agency to safeguard geological repositories. (This brought back memories to me personally as I was on the first U.S. delegation to participate in discussions at the IAEA on geological repository safeguards.)

The third paper, authored by D. Methling, R. Diersch, and H. Spilker of GNB Gesellschaft fur Nulear-Behalter, and titled, CASTOR® X/32S Design and Licensing Status, provides good information on their spent-fuel storage and transportation cask. Finally, the paper by the IAEA's Peter Dyck, Spent Fuel Storage Overviews on a World Wide Basis, summarizes almost every thought one might have regarding the present and future of spent-fuel storage and needs. I feel confident that this article will be referenced quite frequently in future articles.

I am looking forward to the summer issue of the *Journal.* Jim Larrimore, chair of our International Safeguards Division, has gathered numerous papers from different countries addressing integrated safeguards (also know as integrating INFCIRC's 153 and 540). These types of issues somehow serve as benchmarks for the specific topics they address.

I note with interest the new members listed in this issue. If you are a member and know any of the new members, please give them a call and welcome them into the Institute.

We continue to include reports from the chapters, the technical divisions, and some of our standing committees. I hope you find these articles interesting. We are attempting to provide this information to generate interest in these parts of the Institute. Should you have an interest, please do not hesitate to contact the representative of the chapter, division, or committee.

I note that the Awards Program for the 42nd Annual Meeting just arrived in the mail. I would like to remind members that now is the time to nominate a deserving person for the awards that the INMM provides.

Finally, I believe our peer review process continues to run smoothly considering that it is still in its infancy. Assistant Technical Editor Steve Dupree is doing an outstanding job implementing this process. Likewise, those who actually do the reviews are making excellent efforts.

At the INMM Annual Meeting each year, I convene a meeting of those who directly support the *JNMM* (INMM headquarter staff, the assistant technical editor, and the associate editors) to discuss the *JNMM* and its future. Should you have any comments regarding the *Journal,* or have some visionary thoughts about it, please contact one of us. We are listed on the top left on page 1.

As always, should you have any questions or comments, please feel free to contact me.

Dennis L. Mangan
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Ten Nations Participate in Third ESARDA/INMM Workshop

The Third Workshop on Science and Modern Technology for Safeguards examined social and political issues as well as science and engineering research that could have applications to international safeguards, according to Stephen A. Dupree, a member of the INMM International Safeguards Technical Division.

The event, sponsored by the INMM and the European Safeguards Research and Development Association (ESARDA), was hosted jointly by the INMM Japan and Korea chapters and held at the International House of Japan. Ninety-three participants from ten nations and the IAEA, EURATOM, and ABACC attended the four-day event last November in Tokyo.

After an opening plenary session, participants were divided into four working groups. The members of each group concentrated on specific ongoing research and current topics of interest in one of four topical areas:

- Regional Systems and State Systems of Accounting and Control,
- Social and Political Aspects of Safeguards,
- New Safeguards Challenges, and
- Safeguards Technologies.

Invited presentations were made in each working group on research that is underway. These presentations provided a starting point for discussions concerning the possible future relevance or application of the presentation subject to international safeguards. The working groups provided an opportunity for in-depth discussion of the issues raised in the papers, and the potential applicability to the effectiveness and efficiency of international safeguards.

The workshop was very fortunate to have representatives of both ABACC and EURATOM, as well as the IAEA, present. These people have practical experience in the regional and local factors that influence international safeguards.

The discussions, which tended more toward the practical rather than theoretical, were wide ranging and included many issues of current importance to the international safeguards community. Summary presentations by the working group chairs in the closing plenary session provided an opportunity for all participants to learn about the topics and discussions conducted in all working groups and to raise questions regarding each area addressed in the workshop.

Co-chairs for the workshop were Gotthard Stein, Hiroyoshi Kurihara, B.K. Kim, and Cecil Sonnier. Much to our regret, Mr. Kurihara could not participate in the workshop due to illness. His contributions and insights were missed.

Author Submission Guidelines

The *Journal of Nuclear Materials Management* is the official journal of the Institute of Nuclear Materials Management. It is a peer-reviewed, multidisciplinary journal that publishes articles on new developments, innovations, and trends in safeguards and management of nuclear materials. Specific areas of interest include physical protection, material control and accounting, waste management, transportation, nuclear nonproliferation/international safeguards, and arms control and verification. *JNMM* also publishes book reviews, letters to the editor, and editorials.

Submission of Manuscripts: *JNMM* reviews papers for publication with the understanding that the work was not previously published and is not being reviewed for publication elsewhere. Papers may be of any length.

Papers should be submitted in *triplicate*, **including a copy on computer diskette.** Files should be sent as Word or ASCII text files only. Graphic elements must be sent in TIFF format in separate electronic files. Submissions should be directed to:

Dennis Mangan
Technical Editor
Journal of Nuclear Materials Management
60 Revere Drive, Suite 500
Northbrook, IL 60062 USA

Papers are acknowledged upon receipt and are submitted promptly for review and evaluation. Generally, the author(s) is notified within 60 days of submission of the original paper whether the paper is accepted, rejected, or subject to revision.

Format: All papers must include:

- Author(s)' complete name, telephone and fax numbers and E-mail address
- Name and address of the organization where the work was performed
- Abstract
- Camera-ready tables, figures, and photographs in TIFF format only
- Numbered references in the following format:
  1. F.T. Jones and L.K. Chang, "Article Title," *Journal* 47(No. 2):112–118 (1980).
  2. F.T. Jones, *Title of Book*, New York: McMillan Publishing, 1976, pp. 112–118.
- Author(s) biography

Peer Review: Each paper is reviewed by two or more associate editors. Papers are evaluated according to their relevance and significance to nuclear materials safeguards, degree to which they advance knowledge, quality of presentation, soundness of methodology, and appropriateness of conclusions.

Author Review: Accepted manuscripts become the permanent property of INMM and may not be published elsewhere without permission from the managing editor. Authors are responsible for all statements made in their work.

Reprints: Reprints may be ordered at the request and expense of the author. Order forms are available from the Institute's office, 847/480-9573.

IAEA Plans International Safeguards Symposium

The International Atomic Energy Agency, in cooperation with ESARDA and INMM, will conduct a symposium titled "International Safeguards: Verification and Nuclear Material Security," October 29 through November 1, 2001, in Vienna, Austria.

This is the ninth in a series. The last symposium was held in October 1997.

The purpose of the symposium is to foster a broad exchange of information on the international nonproliferation regime, concentrating on concepts, technologies, and experience related to international safeguards.

The Symposium will focus on:

Elements of the international nonproliferation regime, including:

- Nonproliferation policy
- The NPT and Nuclear Weapon-Free Zone Treaties
- IAEA Safeguards Agreements and the Additional Protocol
- Nuclear export control agreements
- The IAEA Action Team: Experience in Iraq.

Implementation of the Strengthened Safeguards System, including:

- Integration of safeguard measures pursuant to comprehensive safeguards agreements and Additional Protocols for optimum effectiveness and efficiency under restrained resources
- Advances in verification technology for declared facilities
- Design information examination, verification, and reverification
- Unattended monitoring systems
- Remote monitoring.

The security of nuclear materials, physical protection measures, and the IAEA's role in relation to illicit trafficking, including:

- The Convention on the Physical Protection of Nuclear Material
- Assistance programs to states for improved nuclear security
- Illicit trafficking in nuclear materials.

International verification in relation to nuclear disarmament, including:

- NPT commitments regarding nuclear disarmament
- IAEA role in fissile material agreements
- The Trilateral Initiative: IAEA verification of weapon-origin and other fissile material in the Russian Federation and the United States

All papers, other than invited review papers, must present original work that has not been published elsewhere. The submission deadline was April 15, 2001.

Anyone wishing to participate in the meeting should complete a participation form and send it as soon as possible to the competent national authority (ministry of foreign affairs or national atomic energy authority) for subsequent transmission to the IAEA. A participant will be accepted only if the participation form is submitted through the competent official authority of an IAEA member state or by an organization invited to participate.

For more information on this symposium, access the IAEA Web site at http://www.iaea.org/worldatom/Meetings/Planned/2001/.

New Look for INMM Web Site on Horizon

INMM officials hope to launch a redesigned INMM Web site in time for the 42nd INMM Annual Meeting July 15-19 in Indian Wells, California. The INMM Executive Committee approved the redesign at its November 2000 meeting. The redesign is being overseen by Communications Chair James R. Griggs and INMM Executive Committee Member-at-Large Cathy Key.

"The redesigned site will be more aesthetically pleasing, provide more up-to-date information, offer improved usability, and serve as an information source on INMM and nuclear materials management," Griggs said.

The new look for the site includes an updated color scheme and improved graphics. The site will be easier to navigate and will help members keep in touch with each other and INMM. Nonmembers will find it easier to learn about the Institute and its goals.

"The appearance of the site will be more in keeping with the new INMM informational brochure," Key said.

Invest in Your Career with the 42nd INMM Annual Meeting

The best investment in your career for the least outlay of time and money is the 42nd Annual Meeting of the Institute of Nuclear Materials Management, July 15-19, at the Renaissance Esmeralda Resort, Indian Wells, California, U.S.A. Plan to be there!

If you are involved in research and development, new concepts, policies, approaches, techniques, equipment, and applications in the fields of nuclear materials management, you won't want to miss this outstanding event.

In addition to the usual INMM meeting activities, we're planning our customary and highly informative technical program—another 300-paper, forty-plus session extravaganza guaranteed to overwhelm you with information you can really use. As an extra treat, our opening plenary speaker will be John A. Gordon, administrator of the National Nuclear Security Administration. Gordon will explain what's happening in this exciting new area.

Attendees of past annual meetings rave about the extraordinary opportunities to meet with many colleagues from around the world, to participate in valuable private meetings, to hear some really outstanding papers, and to gather useful information from other specialty areas that they might not ordinarily encounter. Several attendees have resolved long-standing issues in the professional atmosphere offered at INMM annual meetings. And, as far as training, it's the biggest payoff for management—in less than one week you can learn more than you could absorb with several training courses, and it won't be outdated by the time you get back home.

The INMM Annual Meeting is great for newcomers but absolutely essential for those already established in the field.

Funding constraints for attendees, travel restrictions, and the number of persons permitted by management from one organization to attend a meeting are obstacles to overcome. I'm sure that with some creativity out we can meet those challenges. You could even pay your own way—many of us have done it when totally necessary.

The INMM Annual Meeting is the best bargain you can get in this business today! Just ask anyone who has attended INMM annual meetings in the past.

Also this year, we will have the annual golf tourney, a 3K run, and activities for companions. I hope to see you all there this summer in delightful Indian Wells and I'll be very disappointed if you can't make it. For more information on the INMM Annual Meeting, access the INMM Web site at http://www.inmm.org.

Charles Pietri
Chair, Technical Program Committee
INMM Annual Meeting

42ND INMM ANNUAL MEETING

Reminder

The Institute of Nuclear Materials Management does not accept purchase orders as a form of payment. Please remember to include a check payable to the INMM or a credit card number when registering for the INMM Annual Meeting.

We regret any inconvenience this may cause.

Order Your Copy of the INMM 41st Annual Meeting Proceedings Now

The Proceedings of the 41st Annual Meeting of the Institute of Nuclear Materials Management is available on CD. These proceedings are a valuable reference, containing the complete text of papers presented at the Annual Meeting. Copies are available for $175.

INMM 41ST ANNUAL MEETING

For information, contact:
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Northbrook, Illinois 60062 U.S.A.
Phone: 847/480-9573
Fax: 847/480-9282
E-mail: inmm@inmm.org

Central Chapter

The Central Chapter currently is working on collecting biographies and completing a ballot for a formal election of officers.

Chris A. Pickett
Chair, Central Chapter
Oak Ridge Laboratory
Oak Ridge, Tennessee U.S.A.

Japan Chapter

The 21st Annual Meeting of INMM Japan Chapter was held in Tokyo December 7-8, 2000, under the direction of M. Akiba, meeting program chair. T. Osabe read a welcome statement on behalf of INMM President J.D. Williams. A total of 128 people attended including three guest speakers from Science and Technology Agency, Nuclear Waste Management Organization Japan, and the Crisis Management Office of the Japanese government.

A panel discussion, "Activities of INMM Japan Chapter – Past and Future Role," was held and five past chairmen and vice chairmen participated as panelists.

Nineteen technical papers were presented and discussed at the technical session.

The chapter's fiscal year 2001 business meeting was held December 7, 2000, in Tokyo during the Annual Meeting. The following reports were given and approved:

- The chapter's FY2000 business report and FY2001 business plan.
- The chapter's FY2000 financial report and FY2001 financial budget plan.

The Third INMM/ESARDA Joint Workshop on Science and Modern Technology for Safeguards was held November 13-16, 2000, at International House of Japan in Tokyo. A total of 93 people participated from Australia, Austria, Brazil, Canada, Finland, France, Germany, Italy, Korea, Luxembourg, Japan, Sweden, and the United States. Following the plenary and introductory session, four working groups were formed and sixty technical papers were presented and discussed.

T. Osabe
Secretary, Japan Chapter
Nuclear Material Control Center
Kanagaw, Japan

Northeast Chapter

A new slate of officers was elected for the Northeast Chapter. They are:

President:	E. R. Johnson
Vice President:	Susan E. Pepper
Secretary:	Teri Westerfeldt
Treasurer:	Bruce Moran
Members-at-large	
Two-year term:	Joe Indusi
	Billy J. Cole
One-year term:	Ed Wonder
	John Kerr

The first organizational meeting of the new officers was held February 23, 2001, at the Capitol Hill Club in Washington, D.C. The purpose of this meeting was the official transfer of records, as well as the development of a program for the rest of the year.

E. R. Johnson
President, Northeast Chapter
JAI Corp.
Fairfax, Virginia U.S.A.

Obninsk Chapter

During 2000-2001, Obninsk INMM Chapter members were involved in a number of events in MPC&A fields.

From May 22-26, 2000, the Second International MPC&A Conference was held in State Scientific Center of Russian Federation-Institute for Physics and Power Engineering, Obninsk, Russia. Members of Obninsk Regional Chapter of INMM were heavily involved in organizing and conducting the conference. Several presented papers. About 350 experts from Russia, the United States, the United Kingdom, France, Germany, the International Atomic Energy Agency, Kazakhstan, Georgia, Belorussia, and Egypt, among other countries, attended.

At present the proceedings of the conference, both in Russian and in English, are available on CD and on the IPPE Web site at http://www.ippe.obninsk.ru or http://mpca2000.ippe.obninsk.ru.

During the last couple of years, there has been intensive cooperation between Russia and the United States on the development of reference materials for MPC&A purposes. As a result, the Russian Working Group on RM was established and officially recognized by Minatom. A number of workshops, training courses, and guidance documents on different aspects connected with planning, production, attestation/certification, and use of RM were performed by Russian members of INMM, who are experts in the field.

Gennady Pshakin
President, Obninsk Chapter
IPPE
Obninsk, Russia

Pacific Northwest Chapter

The Pacific Northwest Chapter completed its election of officers and executive committee positions. The new officers are:

President:	Rod Martin
Vice President:	Glenda Ackerman
Secretary/ Treasurer:	George Westsik
Executive Board:	Lupe Ellingson
	Mark Killinger
	Jim Andre
	Brian Smith (Past President)

The chapter had a successful winter dinner meeting on February 20, 2001, that included several visitors from Russia. The speakers were Brian Smith and Gennady Pshakin. Gennady, president of the Obninsk Chapter of the INMM, spoke on the Russian perspective on their recent experiences working on the Trilateral Initiative.

The chapter will continue its longstanding support of the Mid-Columbia Regional Science and Engineering Fair, the Department of Energy-sponsored Science Bowl and other local technical societies and educational activities. The DOE Regional Science Bowl was February 24 and was supported by several members of the INMM.

Brian Smith
Past President, Pacific Northwest Chapter
Deanna Osowski
Secretary/Treasurer, Pacific Northwest Chapter
Pacific Northwest National Laboratory
Richland, Washington U.S.A.

Russian Federation Chapter

A meeting of the Russian Chapter of INMM was held in December 2000. Chapter Chair Y. Volodin delivered a report on activities performed by the Chapter in 1999/2000. Then the following new officials were elected for 2001:

Chair: Eduard F. Kryuchkov
Vice Chair: Alexander V. Izmailov
Secretary: Andrei I. Zobov

Seven new members from six different organizations were admitted to the chapter. The copies of their application forms and their background shall be submitted to the INMM Headquarters in the near future.

The Russian Chapter has twenty-eight members now.

Plans for expansion of chapter activities and for involving in its ranks new specialists on nonproliferation from different regions of Russia were also discussed at the meeting.

Alexander V. Izmailov
Vice Chairman, Russian Chapter

Southwest Regional Chapter

On the evening of January 10, 2001, the Southwest Regional Chapter of the Institute of Nuclear Materials Management enjoyed its First Annual Winter Dinner Meeting at the Kirtland AFB Officers' Club in Albuquerque, New Mexico. More than forty members were able to attend. Highlighting the event was a featured after-dinner presentation by Dr. Roger Hagengruber, senior vice president for national security at Sandia National Laboratories.

Hagengruber traced the history of the Department of Energy and the National Nuclear Security Administration, and outlined hopes and expectations for the future. He stressed the importance of understanding the history of the nuclear programs as we consider the status and prospects for the NNSA. An open and lengthy discussion of related ideas followed the presentation.

The chapter is planning an annual meeting to be held May 4, in Taos, New Mexico. The agenda includes formal presentations in the morning, and breakout sessions covering a variety of topics in the afternoon. Included in the agenda is the annual business meeting of regional chapter, and dinner event. The chapter expects to sponsor a student paper from one of the local universities. On Saturday, the attendees are invited to participate in a river-rafting trip being planned by the executive committee.

Lawrence Kwei
Secretary/Treasurer, Southwest Regional Chapter
Rocky Flats Field Office
Golden, Colorado U.S.A.

International Safeguards Division

INMM will again cooperate with the IAEA on its Symposium on International Safeguards: Verification and Nuclear Material Security, Vienna, Austria, October 29-November 1, 2001. Jim Larrimore, as the INMM point of contact for this cooperation, participated in a planning meeting for the Symposium on January 30 in Vienna; the paper selection meeting is scheduled for May. The IAEA has requested that INMM encourage the submission of papers to the Symposium. Abstracts should be submitted by April 15 through the government authority. The INMM president is expected to represent the INMM in the formalities of the Symposium. INMM has provided some financial support for activities of the Vienna Chapter in conjunction with this Symposium.

The ISD has taken an initiative for a summer 2001 issue of *JNMM* on "Progress towards the integration of INFCIRC/153 and INFCIRC/540." ISD decided to undertake this initiative in view of the IAEA having set the target date of the end of 2001 for completion of the conceptual framework for integrated safeguards. This initiative follows the model of the summer 1998 issue of *JNMM* on "Issues Surrounding the Integration of INFCIRC/153 and INFCIRC/540." That *JNMM* issue contained six articles and a foreword by the IAEA. With commitments from nine countries and organizations and from IAEA, ISD expects that the summer 2001 *JNMM* issue will present a wide array of international views. The papers in this issue will also be presented and discussed at the Annual Meeting.

The next ISD meeting will be held on Monday, May 7, 2001, in Bruges (Brugge), Belgium, in conjunction with the 23rd ESARDA Annual Meeting. Topics for discussion will be announced in the near future.

The ISD will meet on Sunday, July 15, 2001, in Indian Wells, California, in conjunction with the 42nd INMM Annual Meeting, to discuss the papers prepared for the summer 2001 *JNMM*.

An ISD meeting is being planned for Friday, November 2, 2001, in Vienna, Austria, in conjunction with the IAEA Symposium on International Safeguards.

Jim Larrimore, Chair
International Safeguards Division
Del Mar, California U.S.A.

Nonproliferation and Arms Control Division

Thanks to those members of the division who looked at the INMM's Nonproliferation and Arms Control Division charter when we sent it out to the membership in September. Larry Satkowiak and I have incorporated comments and submitted the revised charter to the INMM Executive Committee for its review and approval. At the same time, we have begun to solicit volunteers to fill the officers' positions within the division. Please let us know of your interest in participating in the work of the division. There are many opportunities.

In the nonproliferation world, there have been three major developments to highlight. The Secretary of Energy Advisory Board Task Force on Evaluation of DOE's Nonproliferation Programs with Russia issued its report. Among its recommendations is a greatly increased level of funding for the nonproliferation programs of the DOE.

The report is available at http://www.hr.doe.gov/seab. Much of the expertise on the issues discussed in the report is resident in the members of this division. I look forward to opportunities this year for us to help address some of those issues.

The second major development is the announcement of the launch of the Nuclear Threat Initiative by Ted Turner and former U.S. Senator Sam Nunn. The initiative includes an initial commitment of $250 million over five years for nonproliferation work. As I understand it, the foundation is looking to fund nonproliferation projects that otherwise would not receive funding. Again, the membership of the INMM Nonproliferation and Arms Control Division has the expertise to play a significant role in helping to determine how these funds should be invested.

A third development is the issuance of the Final Draft of the Report of the TOPS Task Force of the Nuclear Energy Research Advisory Committee. TOPS stands for Technological Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems. The report can be found at http://www.ne.doe.gov/nerac/TOPS-Final.pdf. This is another area in which the division's membership has expertise that is relevant.

Last November, the division co-sponsored a session at the winter meeting of the American Nuclear Society/European Nuclear Society. The session was titled "Nonproliferation Policy and Programs: Dealing with Weapons-Usable Nuclear Materials," and included speakers from DOE, INMM, and the ANS.

As is customary, the division will sponsor a number of sessions in this summer's annual meeting of the INMM. In the MPC&A area we expect to have two full sessions and one short session.

We expect to have sessions focusing on dismantlement transparency and authentication. And this year we will sponsor a special roundtable session on nonproliferation education and training. I look forward to seeing as many of members as possible at our division meeting on Sunday, July 15, 2001.

Finally, we have a number of opportunities to influence the future nonproliferation agenda through our efforts at education and outreach. We have considered holding a workshop as early as this spring where we can address some of the issues raised by the SEAB report, and where we can help fill in the blanks for the Turner Foundation on where their investments will do the most good. We think that the subject is timely, and we are anxious to help get the membership engaged in this worthwhile effort.

Steve Mladineo
Chair, Nonproliferation and Arms Control Division
Pacific Northwest National Laboratory
Falls Church, Virginia U.S.A.

Packaging and Transportation Division

Work continues on the 13th International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM 2001).

- The contract between the INMM and MACTEC (DOE contractor controlling the Symposium funding) was completed and signed.
- Identification and solicitation of attendees for the paper review is complete.
- More than 200 paper abstracts were received.
- Identification of potential keynote speakers has begun.
- An exhibitor prospectus and exhibit contract were finalized and distributed to potential exhibitors.

Billy Cole
Chair, Packaging and Transportation Division
JAI Corp.
Fairfax, Virginia U.S.A.

Physical Protection Division

The Physical Protection Technical Division is planning a workshop for fall 2001. The target dates are September 18-20, providing a reasonable gap after the INMM Annual Meeting. The tentative location for the meeting is Cincinnati, Ohio. This locale was selected for two reasons. One is that Cincinnati may be accessed by direct flight from many locations in the United States; the other is its proximity to the Portsmouth Gaseous Diffusion Plant. The intent is to provide a practical look at the plant's security applications.

Topics for the workshop will include the integration of operational considerations into overall security assessments. The plan is to integrate the concept of operations, MC&A, and physical protection. A tentative title for the workshop is "ROI—Value-Added Integrated Safeguards and Security." We expect the workshop to attract 20-40 participants. L. David Lambert is the coordinator for this event, and he is working with INMM Headquarters to secure a location.

The Physical Protection Technical Division has been working very hard to solicit good papers for the 2001 Annual Meeting. I have identified key personnel to help with developing sessions in specific areas of physical protection. Rebecca Horton, Sandia National Laboratories, is working to identify papers in access control and contraband detection. Laura Thomas, Oak Ridge National Laboratory, is collecting papers on information security. Boris Starr, Sandia National Laboratories, is screening papers on intrusion detection. We expect to have four or five strong sessions on physical protection at the Annual Meeting.

Stephen Ortiz
Chair, Physical Protection Division
Sandia National Laboratories
Albuquerque, New Mexico U.S.A.

Waste Management Division

The following summarizes the activities of the Waste Management Division for the period November 2000 through February 2001:

The INMM Spent Fuel Management Seminar XVIII was held January 10-12, 2001, at the Willard Inter-Continental Hotel in Washington, D.C. Approximately 135 people attended the seminar. Attendees included representatives of utilities, vendors, government and international agencies, regulators, national laboratories, consultants, and the press. The representation was international, including the attendees from Austria, France, Germany, Japan, and Spain.

The meeting facilities at the Willard were very good and the hotel staff was extremely helpful. In view of this, INMM requested rates from the Willard for holding the 2002 and 2003 seminars. A quote has been received and is under advisement. The dates we would like to hold the seminar in 2002 are not available, and restrictions placed on the next best dates might be unacceptable. In view of these potential problems, INMM will be asked to obtain rates from the Renaissance Mayflower (site of the January 2000 seminar).

E. R. Johnson
Chair, Waste Management Division
JAI Corp.
Fairfax, Virginia U.S.A.

Membership Committee Report

We have just completed the INMM membership renewal period. The membership status of the Institute of Nuclear Materials Management as of February 21, 2001, is as follows:

595	Regular Members
81	Senior Members
19	Fellow Members
7	Fellow Emeritus Members
17	Emeritus Members
3	Senior Emeritus Members
2	Student Members
25	Sustaining Members
1	Honorary Member

Total Membership: 750

The goal of the Membership Committee is to provide quality service to INMM members. The Membership Committee is currently comprised of Nancy Jo Nicholas (chair), Jill Cooley, Bob Curl, Vince DeVito, Al Garrett, Michelle Kazanova, Larry Kwei, Bruce Moran, Takeshi Osabe, Don Six and Scott Vance.

Review of the INMM Membership Directory for 2001 by the Membership Committee will begin soon. We hope to go to print with this year's directory in early April and mail it to the membership by mid-April.

The Membership Committee is working with *JNMM* editors to expand the page containing names and contact information of new members to a whole page on "Member News"—short pieces about the careers of INMM members. Please let us know about promotions, awards, retirements, and other career news so we can highlight what's going on with our membership, and send a photo, if available.

The Membership Committee is also preparing for our annual review of applicants for Senior Membership status. Applications were due to INMM headquarters April 1. The Membership Committee reviews the applications and makes a recommendation to the Executive Committee. New seniors are recognized in July during the Annual Meeting.

Nancy Jo Nicholas
Membership Committee Chair
Los Alamos National Laboratory
Los Alamos, New Mexico U.S.A.

Government Liaison Committee

At the November Executive Committee meeting, we agreed to consider selecting some speakers for the closing plenary from the same pool of candidates being considered for the opening plenary speaker. In this regard, we have been following progress in selection of the opening plenary speaker.

In January, we also heard from several members of the Technical Program Committee that INMM may organize a session about the Trilateral Initiative for monitoring progress in dismantlement of nuclear weapons with high-level participants, possibly at the ministry advisory level, from the United States, Russian Federation, and the IAEA. The committee chair has suggested that the closing plenary would be an appropriate venue for such a session.

Some time ago, Tohru Haginoya, a GLC member, offered to distribute news and fact sheets about the Japanese nuclear industry, initially to a limited group of INMM members. As agreed previously, the names and e-mail addresses of the members of the Executive Committee and the chairs of the standing committees and technical divisions were sent to Mr. Haginoya.

James Lemley
Chair, Goverment Liaison Committee
Brookhaven National Laboratory
Upton, New York U.S.A.

R&D Program for Interim Storage of Spent Fuel in CRIEPI

A. Okumura, K. Shirai, and T. Saegusa
Central Research Institute of Electric Power Industry

Abstract

Japan has consistently promoted the closed nuclear fuel cycle to achieve a stable supply of energy for the future and to minimize environmental impact. A domestic commercial reprocessing plant, with a capacity of 800tU/a, is under construction. The aim is to be operational by 2005. However, the rate of spent fuel arising in Japan will be more than the reprocessing plant's capacity. It is estimated that in 2010 interim-storage facilities of spent fuel, whose capacity will be about 7,700tU, will have to be constructed as away-from-reactor sites. In 1999, a revised bill, which specifies operational and regulatory conditions for interim-storage facilities away-from-reactor sites, passed the Diet and it was put into force in June 2000. At present, an advisory committee of the Ministry of International Trade and Industry (MITI) of the Japanese government is now discussing AFR storage. A private entity, which will be licensed by the government, will be able to construct the interim-storage facilities AFR sites.

Research and development on spent-fuel storage has been carried out by the government and the electric power companies, particularly aiming at the realization of dry-storage away from reactor in 2010 (Shirai, 1999). CRIEPI, the Central Research Institute of the Electric Power Industry, has been involved in R&D programs on spent-fuel storage technologies and long-term storage on behalf of MITI.

Introduction

Japan has scarce energy resources and is dependent on imports of foreign resources for most of its energy needs. Therefore, Japan has made efforts to use nuclear power since the mid-1950s, carrying out research and development, and promoting commercialization of peaceful use of nuclear energy. Since its initial stage, the development and utilization program has consistently called for recycling nuclear fuel.

Research and development on spent-fuel dry-storage technology has been carried out mainly by CRIEPI under contracts with MITI and the Science and Technology Agency (STA) of the Japanese government.

In a study program completed in 1996, safety and cost evaluations for various types of dry-storage systems applicable to the storage at reactor sites were carried out for spent high-burnup fuel and MOX fuel. Especially, for metal cask storage technology, the safety of the total system was confirmed. Some of the results were incorporated in a safety review guide, *Dry Cask Storage of Spent Fuel in Nuclear Power Plants* (Nuclear Safety Commission, 1992). They were used in the licensing process of the first dry-cask storage at the reactor site of the Fukushima-Dai-Ichi nuclear power station of the Tokyo Electric Power Co.

From April 1992 to March 1997, a study program of advanced spent-fuel storage technology had been executed at CRIEPI. This five-year program relates to the dry storage of the high-burnup and MOX spent fuels, which will be the main object in the next stage of storage strategy. The outlines had been introduced at the INMM Spent Fuel Management Seminar XVII in 2000.

This paper introduces the perspective of CRIEPI on the current status of a new R&D program of spent-fuel storage technology and long-term storage consigned by MITI from April 1997.

New R&D Program for Interim Storage (FY 1997-2003)

Beginning in March 1997, the MITI, the STA, and electric power companies studied the introduction of the interim storage of spent-fuel AFR sites, and issued a report in March 1998. The Subcommittee for Nuclear Energy of the Council for Conventional Energy Policy, an advisory body for MITI, also issued a similar report in June 1998 (Subcommittee for Nuclear Energy, 1998). The report pointed out the necessity of introducing such interim-storage facilities by around 2010.

In 1997, a new study program of verification tests for interim storage of recycle fuel resource was started. This is mainly related to concrete modular storage technology, such as a horizontal concrete silo and a concrete cask. A concrete modular storage system is considered to have an economic advantage. To propose "safety standards for concrete modular structures, systems, and components," the following studies, related to especially concrete cask, are now in progress (Saegusa, 2000, Shirai, 2000).

I. For concrete material and structures
   A. Long-term durability of concrete material (carbonation and salt damage)
   B. Dynamic strength for concrete material under high temperature and in case of accidents
   C. Characteristics of heat transfer and cracking due to thermal stress
   D. Shielding performance of concrete structures
II. For metal canister
   A. Impact and corrosion resistance of multipurpose canister with the welded components
III. For spent fuel
   A. Development of nondestructive monitoring method
   B. Characteristics and long-term performance of high-burnup and MOX spent fuel
IV. Demonstration program for qualification of concrete cask performance
   A. Basic design of Japanese type concrete cask
   B. Fabrication of full-scale concrete cask
   C. Demonstration tests (e.g., heat removal, impact, and seismic tests)
   D. Safety Analysis

The following sections outline the current status of this new R&D program.

I. Study for concrete material and structures

As the concrete modular structures for the dry storage of spent fuel may be used under high temperature and severe radiological conditions, reinforced concrete components important to safety must be designed, fabricated, erected, and tested to withstand the effects of natural phenomena and accident condition.

Figure 1 shows the schematic of the performance of reinforced concrete components during dry storage.

A. Long-term durability of concrete material

If the storage site is located near the seashore, the reinforced concrete components and structures must withstand damage due to salt and carbonation under high temperature.

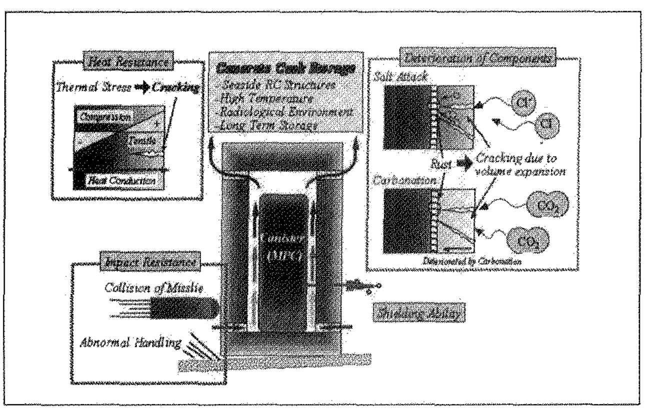


Figure 1. Schematic of reinforced concrete components during dry storage

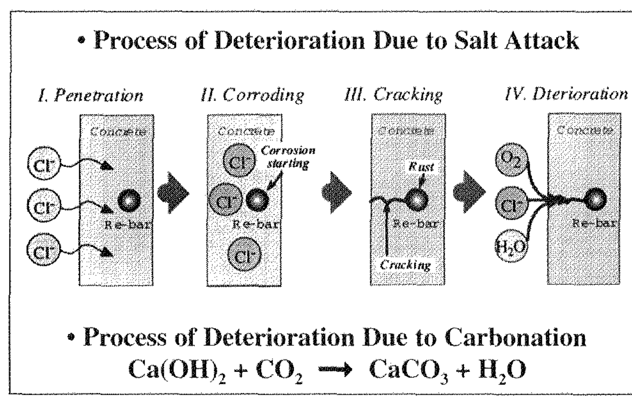
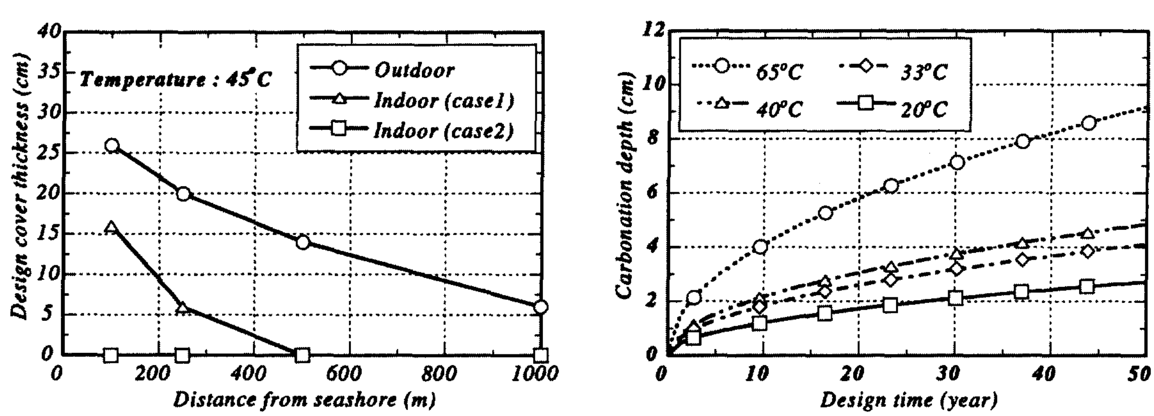


Figure 2. Process of deterioraton due to salt attack and carbonation



(Design curve for salt attack) (Design curve for carbonation)

Figure 3. Example of draft design curve for carbonation and salt attack

Table 1. Design allowable concrete temperatures (AIJ, 1978)

State	Temperature Limitations (°C)	
	General area	Local area
Normal Operation	65	90[1]
Accident	175	3,502[2]

Note: 1) such as around a penetration
2) steam jets in the event of a pipe failure

Figure 2 shows process of deterioration due to salt and carbonation. Salt leads to the cracking of the reinforced concrete components because corrosion and volume expansion of the steel bars occur due to the reaction with the penetrating chloride ion. Carbonation can be defined as the reaction between $Ca(OH)_2$ in the concrete and CO_2 in the atmosphere. As pH value in the concrete decreases due to carbonation, the corrosion and volume expansion of the steel bars will also possibly damage the reinforced concrete components.

It can be noted that temperature considerably affects this kind of damage. However, there are very few examination examples to make clear the influence of the temperature. To obtain the basic material properties at high temperature and evaluate the long-term durability of concrete materials, the following tests are now in progress.

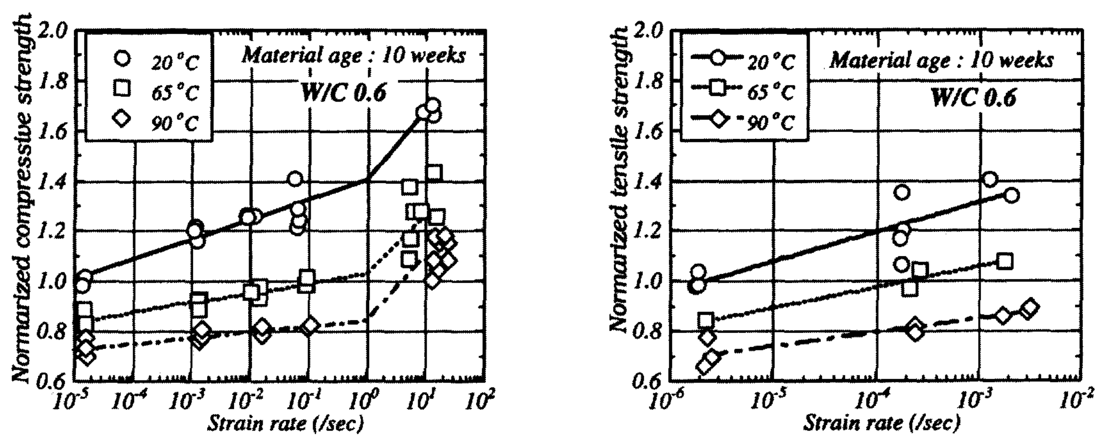
1. Chloride ion diffusion test and carbonation accelerated test in concrete under high temperature.
2. Corrosion tests of steel bars in concrete under high temperature by pre-mixed salt solution.

Figure 3 shows the draft design curve for carbonation and salt damage. At high temperature, the deterioration due to salt attack and carbonation seems to be highly accelerated.

B. Dynamic strength for concrete material under high temperature and in case of accidents

In the Japanese Design Code for Concrete Rector Vessels (Architectural Institute of Japan, 1978), design-allowable concrete temperatures are specified as shown in Table 1. Higher temperature than given in this table may be allowed if evidence can be provided which verifies that the increased temperature does not cause deterioration of the concrete. This design principle is introduced from ASME Section III, Division 2.

In the Japanese design code (*AIJ,* 1978), there is a notice that "Material characteristics must be provided considering the reduction in strength. For example, the concrete strength



(Compressive strength) **(Tensile strength)**

Figure 4. Example of the material tests under high temperature (Normal concrete: W/C 60%)

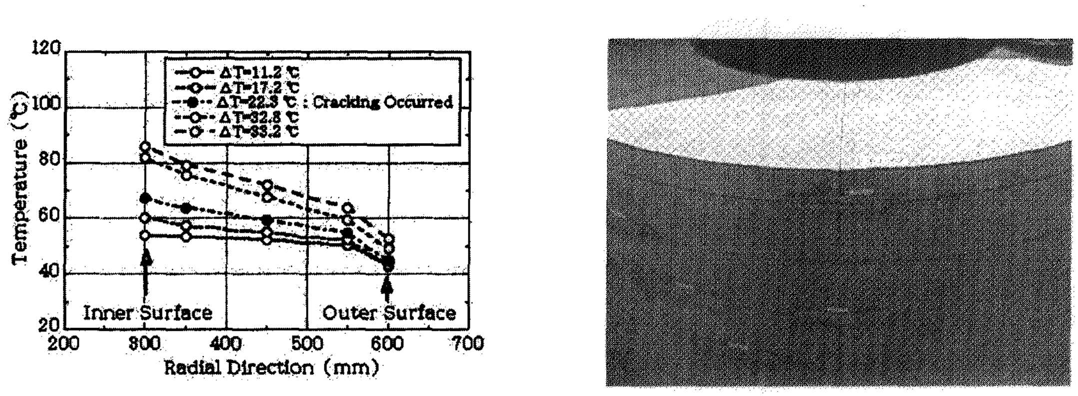


Figure 5. Example of the heat transfer test.

decreases under high temperature below 110°C, and around 50°C the minimum value exists. However, there are very few test examples to make clear the influence of the temperature." To obtain the dynamic material properties at high temperature, the following tests are now in progress:

1. Dynamic and impact test of concrete material under high temperature
2. Fracture toughness test of concrete material under high temperature

Figure 4 shows the example of the material tests under high temperature with normal concrete. It is found that reduction in strength of the normal concrete will not be avoidable at high temperature.

C. Characteristics of the heat transfer and the cracking due to temperature stress

If the design-allowable concrete temperatures as shown in Table I are considered, temperature stresses are one of primary factors governing the design of reinforced concrete hollow cylindrical structures subjected to temperature gradients such as concrete cask bodies. Therefore, a rational evaluation of safety against such stresses must be established.

To investigate the conditions in which cracks due to temperature gradients initiate in reinforced concrete structures, the heat transfer tests with several scale concrete structures are now underway.

Figure 5 shows an example of the heat transfer test.

D. Shielding performance of concrete structures

To evaluate the shielding ability of the concrete cask during the storage period, the following studies are now in progress.

1. Evaluation of dose rate due to streaming thorough air inlet and outlet

It is very important to estimate dose rate due to streaming thorough air inlet and outlet. A numerical code to evaluate shielding performance, Monte-Carlo calculation method, is used for analysis. The experimental program concerning streaming with the scale-model of cooling ducts is being planned to investigate the accuracy of the Monte-Carlo calculation scheme and model.

2. Effects of cracking of the concrete body on shielding performance

Due to cracking as a result of thermal stress, shielding performance will depend on the cracking width. However, there are no design criteria which gives an allowable crack size for shielding. Therefore, shielding analyses with slit model simulating a concrete crack with Monte-Carlo calculation code have been executed. Figure 6 shows analytical result with the slit model. It seems that there is not so considerable influence of crack width up to 0.4mm on shielding ability.

3. Evaluation of dose rate at site boundary

In the conventional design for evaluation of dose rate at site boundary of the spent-fuel storage facility, the point source or uniform volume source has been conservatively considered without taking account of any mutual shielding effect between the contiguous casks. If the real geometry of the cask storage area is considered, the mitigation of the dose rate at site boundary can be expected because of this effect. Therefore, to propose the rational design method, the calculation model considering the real geometry of the cask storage area is being developed with Monte-Carlo calculation code. Figure 7 shows an example of analytical results. It seems that the additional shielding effect due to the mutual shielding effect will be expected.

II. Study for metal canisters

For the welding structure of metal canisters, the development of the methodology to evaluate the integrity of the canister's weld parts is very important since the integrity cannot be evaluated only by X-ray examination and annealing heat treatment. Therefore, the fracture toughness test and the corrosion test relating to stress corrosion cracking of canister's weld part are planned.

Figure 8 shows the schematic of the performance of metal canister components during dry storage.

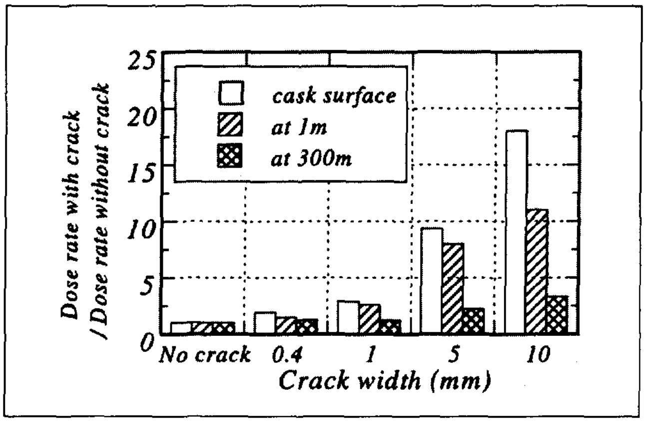


Figure 6. Influence of crack width on shielding performance

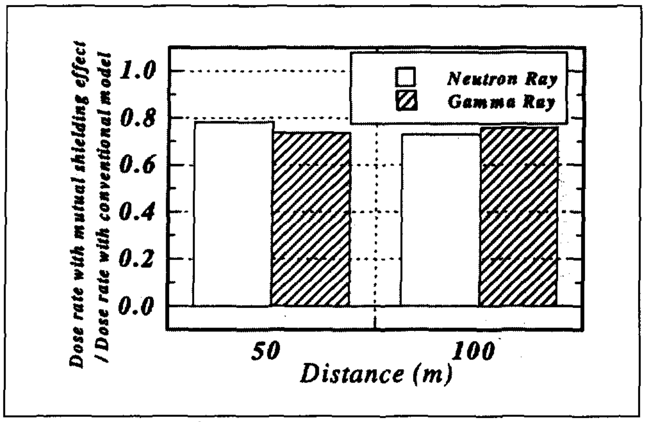


Figure 7. Mutual shielding effect

III. Study for high-burnup and MOX spent fuel

A. Characteristics of high burnup and MOX spent fuel

The high-burnup PWR/BWR spent fuel and PWR MOX spent fuel irradiated in commercial reactors were used in this study. Table II shows the summary of the spent fuel.

Several post-irradiation examinations were executed to evaluate the neutron/gamma source of spent fuel and the introduction of the burnup credit into the interim storage.

- Neutron emission and gamma ray source distribution along the fuel rod
- Gamma ray source distribution in the radial direction of the pellet
- Chemical isotopic analyses for the nuclide composition of actinides and fission products of the pellet

The following examinations were also performed to evaluate the fuel characteristics during irradiation and storage.

- SEM/TEM observation
- Annealing test for the fission gas release behavior
- EPMA observation of U, Pu, Xe, Cs, and Nd along pellet radius

In the chemical isotopic analyses, the burnup was estimated by the amount of heavy metal and ^{148}Nd. The analysis results were compared with ones of ORIGEN2/82 calculation. The C/E (ratio of calculation and experimental value) of actinides is shown in Figure 9 for high burnup PWR fuel (Sasahara, 1996a, 1996b, 1997). The differences between calculation and experimental value in isotopic composition were considered in reactivity bias of the burnup credit (Matsumura, 1998).

In the annealing test, the pellets obtained from high burnup PWR fuel and PWR MOX fuel were annealed up to 1,700°C with staircase. The fuel temperature at vacuum condition shall be around 500°C in exchanging the inner gas in cask and canister to helium gas during spent fuel loading. According to the test results, less than 2 percent fission gas will be released up to 500°C, however this additional fission gas release will have a lesser impact on fuel rod pressure.

B. Long-term performance of high-burnup and MOX spent fuel

BWR MOX spent fuel, of which burnup is about 20GWd/t, was used to evaluate the integrity of spent fuel during twenty-year wet and dry storage. The following examinations were performed:

- Visual inspection of the pre/post-wet-storage fuel
- Puncture test of the pre/post-wet-storage fuel
- Atmospheric gas analyses of the pre/post-dry-storage fuel
- Ceramographic examinations of the pre/post-dry-storage fuel

According to the preliminary observation, there is no marked difference after storage of these low-burnup rods. However, further examination may be required to make clear the integrity of MOX spent fuel during long-term storage.

IV. Demonstration program for qualification of concrete cask performance

Based on the prescribed test results, the demonstration program for qualification of concrete cask performance is being started. Some important test issues are as follows.

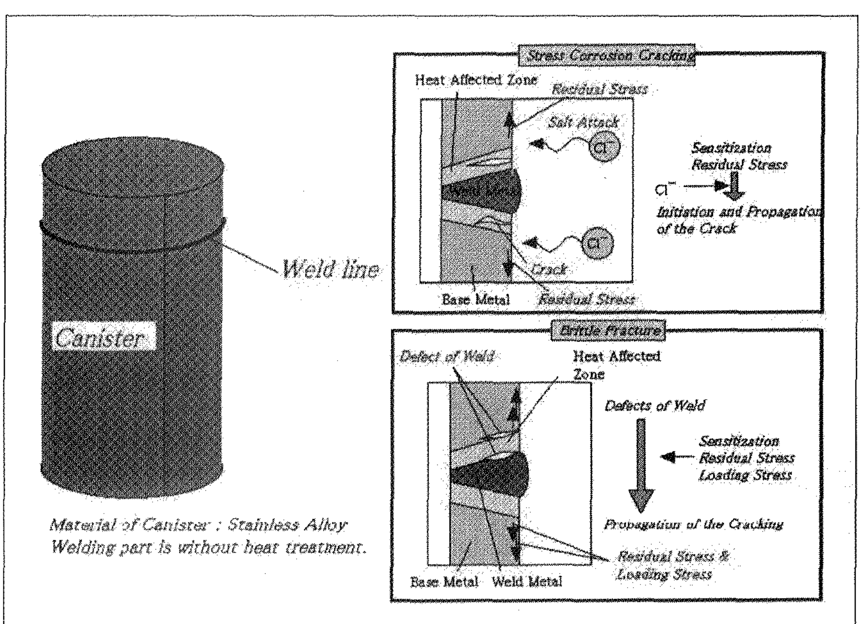


Figure 8. Schematic of the performance of metal canister components during dry storage.

A. Basic design of Japanese type concrete cask

Development of two types of concrete cask as shown in Figure 10, reinforced-concrete type and steel-layered concrete type, to store the high-burnup spent fuel are now in progress.

B. Fabrication of full-scale concrete cask

Two types of full-scale concrete casks and multipurpose canisters will be fabricated to apply to the demonstration tests.

C. Demonstration tests

Heat removal tests of concrete casks considering the normal, abnormal, and accidental events, and the metal canister are planned. Seismic test with scale-model cask and streaming test with the air inlet components are also planned.

D. Safety Analysis

To contribute to "safety standards for concrete modular structures, systems, compo-

Table II. Burnup/enrichment of Spent Fuel

Fuel type	Rod average burnup	Enrichment/Puf enrichment
High-burnup PWR	60 GWd/t	3.8
High-burnup BWR	56 GWd/t	3.5
PWR-MOX2	45 GWd/t	3.5

nents," safety analysis will be performed using the results obtained in demonstration tests.

SUMMARY

The Japanese policy on nuclear fuel cycle is clear, and we intend to guarantee future energy security by steadily carrying forward research and development efforts aimed at future commercial commissioning of nuclear fuel recycling. Since spent fuel discharged from nuclear power stations is considered to be a valuable quasi-domestic energy resource containing plutonium and uranium, the amount of it in excess of domestic reprocessing capacity will be appropriately stored as an energy stockpile until such time as it can be reprocessed.

A domestic commercial reprocessing plant that will be operating by 2005 is now under construction. Interim storage facilities of spent fuel away-from-reactor sites will be constructed by around 2010.

Based upon Japan's nuclear fuel cycle policy stated in the long-term program, CRIEPI steadily continues to develop safe and economical technologies for dry-storage technology on spent fuel.

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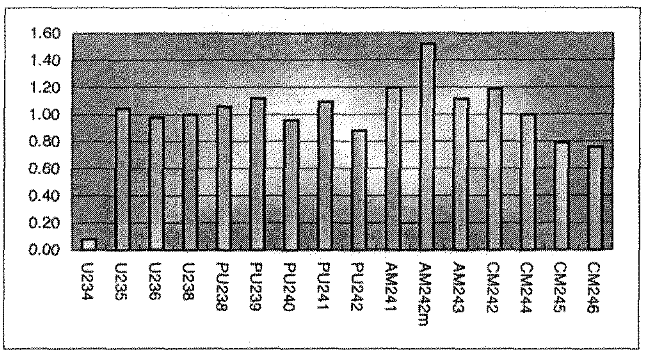

Figure 9. Chemical analysis and ORIGEN2/82 results for high burnup PWR fuel (65GWd/t at local point)

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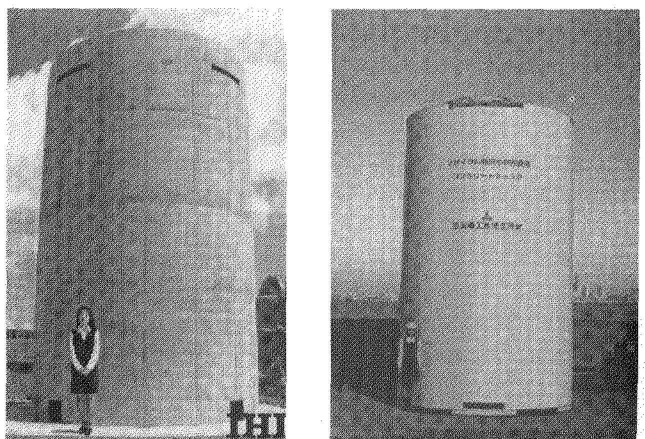

[Reinforced concrete type: RC cask] **[Steel-layered concrete type: SC cask]**

Figure 10. Prototype of concrete cask

International Atomic Energy Agency Safeguards Impacts on Geological Repositories

Bruce W. Moran
U.S. Nuclear Regulatory Commission
Washington, D.C.

Background

Safeguards Approach Development

The International Atomic Energy Agency began technical studies on safeguards measures necessary to effectively and efficiently detect the diversion of nuclear material from geological repositories in 1988. The IAEA held consultants' meetings in 1991 and 1995, and an advisory group meeting in 1997 to support its development of safeguards policy and safeguards approaches for geological repositories.[1] The IAEA issued its policy paper on geological repository safeguards in 1997. The IAEA convened the International Program for the Development of Safeguards on the Final Disposal of Spent Fuel in Geological Repositories (SAGOR Program) in 1994.[2] The SAGOR Program issued its final report recommending safeguards approaches for generic spent fuel conditioning plants, operating repositories, and closed repositories in 1998. After completion of the SAGOR program in 1999, the IAEA established the Geological Repository Safeguards Experts Group to provide guidance on safeguards approach implementation issues.

Spent-Fuel Disposal Facility Status

Facilities for the final disposal of spent fuel in geological repositories are being designed, built, and prepared for operation. The German spent-fuel conditioning plant has been built and licensed, and a safeguards approach for it has been negotiated with both EURATOM and the IAEA. The facility is awaiting German government approval before it begins repackaging spent fuel from storage casks into the final disposal casks. Construction of spent-fuel conditioning plants is scheduled to begin in the United States in 2005, in Sweden in 2007, and in Finland in 2010. Finland has submitted its conditioning facility design documentation to the IAEA and EURATOM and began discussion on the safeguards approach this year.

Repository site selection is scheduled to occur in the United States and Finland in 2001 and in Sweden in 2005. Completion of site selection actions in Germany has been postponed for three to ten years. Excavations at candidate repository sites have already occurred in Germany and the United States, and are scheduled to begin in Finland as early as 2003 and in Sweden in 2009. Geological repository programs in other countries (e.g., Canada, Belgium, Switzerland, Hungary, Spain, France, and the United Kingdom) are either further in the future or the countries have not yet officially declared their schedules. Repository conceptual designs have been prepared in several countries and will be submitted for licensing consideration, beginning in the next few years with the U.S. Yucca Mountain Repository application, which is scheduled to be submitted in 2002 (assuming site selection).

State-IAEA Interactions

The IAEA places particular importance on early information exchanges on newly declared facilities. This permits the IAEA, the state, and the facility operator to discuss the safeguards approach and necessary safeguards measures such that the impact of safeguards on facility operations can be minimized and the effectiveness of safeguards enhanced. Early cooperation also reduces the cost of safeguards implementation on both the operator and the IAEA. IAEA safeguards policy with respect to geological repositories states that the pre-operational phase of the geological repository safeguards approach begins with a decision by the state to construct a geological repository. Soon after that time, consultations are to begin with the state to agree on and begin to implement safeguards measures.

Safeguards Implementation Status

At this time, the IAEA has not completed its studies in preparation for the application of safeguards to a geological repository program. As a result of the discovery in 1991 of undeclared nuclear facilities and a nuclear weapons program in Iraq, a Nonproliferation Treaty signatory state, the IAEA's safeguards priority was shifted from enhancing traditional safeguards to strengthening the IAEA's safeguards program to enhance the detection of undeclared material, facilities, and activities. This new safeguards responsibility, as well as the responsibility to apply safeguards to a constantly increasing number of facilities, has to be implemented by the IAEA during the second decade of zero-real-growth of the agency budget. In 1993, the IAEA initiated its Strengthened Safeguards Program and, in 1995, began implementing new safeguards measures to strengthen the effectiveness and efficiency of IAEA safeguards within the existing statutory authorities. In 1997, negotiation of new statutory authorities was completed and the Model Additional Protocol to Safeguards Agreements was approved by the IAEA Board of Governors.[3] Since 1997, the IAEA member states have begun negotiating, signing, ratifying, and bringing into force Additional Protocols to their Safeguards Agreements. Once the Additional Protocol has been brought into force in a state, the IAEA has the right to collect information on the nuclear, and nuclear-related activities in the state, and to seek access to locations to verify the accuracy of the state declarations or to resolve questions on the information.

In 1997, the IAEA began to evaluate how to integrate the new strengthening measures with the traditional measures to produce a more effective and efficient safeguards system. For those states with comprehensive safeguards agreements with the IAEA and an Additional Protocol in force, and in which the IAEA has drawn conclusions on the absence of diversion of nuclear material and on the absence of undeclared nuclear material and activities, reductions in the current level of safeguards verification effort will be permitted on less sensitive nuclear materials (e.g., depleted, natural, and low-enriched uranium and irradiated fuel). The principles for implementing the integrated-safeguards system are still being developed as well as proposals for implementing integrated safeguards at different facility types. The IAEA has not yet begun to assess how integrated safeguards should be implemented for geological repositories.

Because many view geological repositories as not being a current safeguards problem, but an issue that is still a decade away, geological repository safeguards has been assigned a lower priority than other safeguards issues. The Geological Repository Safeguards Experts' Group has maintained pressure on the IAEA to make decisions regarding implementation of geological repository safeguards. At the December 2000 Experts' Group meeting, the IAEA was asked to review credible diversion paths for geological repositories and to define the measures composing the safeguards approach for geological repositories. This information will permit the Experts' Group to assess safeguards techniques that could meet the technical needs of the safeguards approach.

Geological Repository Safeguards

Safeguards Policy

The IAEA's geological repository safeguards policy contains the following concepts:

1. Spent fuel disposed in geological repositories is subject to safeguards in accordance with the IAEA's safeguards agreement with the state. Safeguards on the spent fuel will be maintained after the repository has been back filled and sealed, for as long as the safeguards agreement remains in force.
2. The safeguards system will be based on (a) verification of the repository design information during design, construction, and operation; (b) verification of receipts and flows to ensure that no undeclared removal of nuclear material occurs; and (c) maintenance of continuity of knowledge on the nuclear material content.
3. Safeguards requirements should be integrated into the repository design at an early stage to establish functional, nonintrusive, and cost-effective safeguards measures.
4. Because emplaced spent fuel cannot be reverified, sufficient redundancy, diversity, and robustness should be incorporated into the safeguards approach and adequate maintenance measures should be applied to avoid system failure and to ensure continuity of knowledge.
5. The safeguards measures should be designed to provide verification of the nuclear material contents of incoming spent-fuel containers and continuity of knowledge of the spent-fuel inventory. As far as practical, the safeguards measures should function in automated, remote control, and remote data transmission modes.
6. Design information verification should confirm the design of the geological repository, including its surface facilities, and detect any undeclared activities, both in the repository and in its vicinity. The excavation areas should be verified to be as declared and that there are no undeclared excavations. The safeguards approach should provide assurance of the absence of undeclared underground reprocessing and assurance of no undeclared operational capability underground that could facilitate and mask transfer of spent fuel to other containers.
7. In the pre-operational phase (i.e., construction phase preceding receipt of first disposal container), the IAEA, in collaboration with the state, should establish all safeguards-pertinent information about the original undisturbed site. Design information verification should be conducted on all excavations and facilities.
8. During the operational phase of the geological repos-

itory, the safeguards system should verify transfers, flows, and inventory of spent-fuel disposal containers through the application of elements of containment and surveillance, monitoring, nondestructive assay, and design information verification.

9. The safeguards measures should verify that the containers declared to be emplaced are actually emplaced; that the containers are not opened, switched or removed; and that there are no undeclared operations that could change the amount of nuclear material in any disposal container.
10. During the post-closure phase, the safeguards system should consist of sufficient surface monitoring measures to provide assurance of no undeclared attempts to obtain access to nuclear material. These measures may include information analysis, visual observation, remote surveillance (e.g., satellite), geophysical monitoring, and environmental techniques.

Safeguards Approach

At a spent-fuel conditioning plant, the safeguards approach will be based on materials accountancy combined with design information verification and appropriate containment, surveillance, and monitoring systems. The safeguards approach may use techniques such as nondestructive assay, tamper-indicating seals, surveillance cameras, motion sensors, radiation monitors, and weight monitors. These measures would be applied in both the cask handling areas and in the hot cells. Input data will be based on shipper's data and the materials accountancy based on item accounting. If the spent fuel was verified by the IAEA before shipment to the conditioning plant and continuity of knowledge has been maintained, the safeguards objective can be met through maintaining continuity of knowledge in the conditioning plant (through containment, surveillance, and monitoring) and ensuring that the disposal container leaves the facility under effective containment and surveillance. If continuity of knowledge is lost or if spent-fuel assembly consolidation occurs, confirmatory verification measurements may be required.

For operating repositories, the recommended safeguards approach, at surface facilities (including spent-fuel handling facilities and hoist and ventilation shafts), is based on materials accountancy combined with design information verification and appropriate containment, surveillance, and monitoring systems. This aspect of the safeguards approach may use techniques such as nondestructive assay, tamper-indicating seals, surveillance cameras, motion sensors, radiation monitors, and weight monitors. Material accountancy is to be based primarily on maintaining continuity of knowledge at all credible access routes to the underground repository. Below ground, the primary safeguards measure is design information verification, with other measures, if needed. Design information verification conducted from above ground and below ground will include visual observation and geophysical techniques (including active and passive seismic, active and passive electromagnetic, and ground-penetrating radar). Environmental sampling through continuous air sampling and samples for trace analysis will be used to detect opening of casks and reprocessing of spent fuel.

For a closed repository, the safeguards approach will be based on surface and aerial verification activities. Reverification of nuclear material in a closed repository will not be possible. In addition, safety of the repository must not be jeopardized. Safeguards measures that could be used to detect undeclared activities include satellite imagery, aerial reconnaissance, inspector observation of the surface area, and geophysical monitoring (including active and passive seismic techniques and active and passive electromagnetic techniques).

In June 2000, a meeting of geophysical experts on geophysical techniques was held to determine the feasibility of using geophysical techniques for safeguarding geophysical repositories.[4] The experts identified that no single geophysical technique can meet all of the potential geophysical monitoring needs or will be applicable in all geological media. Each technique has limitations in terms of range and resolution in different media. Thus, use of the techniques must be evaluated for each proposed geological repository.

Summary and Conclusions

The IAEA has an approved safeguards policy for implementing safeguards at geological repositories. Safeguards approaches have been recommended for generic conditioning plants and operating and closed geological repositories by the SAGOR Program and a 1997 Advisory Group Meeting. Member State Support Program tasks have been conducted to develop and evaluate technologies that could be used to implement the safeguards approaches. Completion of these evaluations requires input from the IAEA on acceptance criteria.

The IAEA has placed a low priority on resolving geological repository safeguards issues because of the higher priority of addressing and implementing strengthened safeguards and integrated safeguards systems. The Geological Repository Safeguards Experts' Group has been encouraging IAEA management to take action on geological repository safeguards implementation. IAEA geological repository safeguards policy indicates that the IAEA should define and approve a safeguards approach for geological repositories, conclude discussions on a site-specific safeguards approach in Finland, and implement the pre-operational safeguards approach within the next three years. If the Yucca Mountain facility is designated as the U.S. geological repository, the facility will be added to the U.S. list of facilities eligible for IAEA safeguards. After the facility is added to the list, the IAEA will need to decide whether to select the repository for implementation of IAEA safeguards.

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CASTOR® X/32 S Design and Licensing Status

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1. Introduction

The CASTOR® X/32 S cask is designed for storage and transportation of thirty-two spent nuclear fuel assemblies from U.S. PWRs. The criteria for acceptance of spent-fuel assemblies for storage in the CASTOR® X/32 depends on their enrichment, burnup and cooling time. These parameters are used to determine the source terms for criticality, shielding, and thermal analyses. On the basis of these analyses, the final layout of the cask was performed.

2. Design Criteria

The CASTOR® X/32 S is designed to store most types of PWR fuel assemblies used in the commercial U.S. nuclear power reactors. These are generally supplied by Westinghouse, ANF, and ABB Combustion Engineering. Therefore, the cell openings and length in the fuel basket have been sized to accommodate thirty-two intact UO_2 PWR 14x14, 15x15 or 17x17 Zircaloy-clad fuel assemblies, made by the above-mentioned manufactures, without or with burnable poison rod assemblies (BPRAs). Damaged fuel assemblies or fuel debris are not permitted for storage.

A summary of design-basis fuel criteria is shown in Table 1.

3. Description of the CASTOR® X/32 S Design

The CASTOR® X/32 S concept destined for the U.S. market has been developed in two different versions. Version I is the original CASTOR® cask concept with a carbon-steel body and with a bolted primary and secondary lid and with pressure monitoring of the interlid space during storage. Version II is similar to the Version I but has a welded secondary lid, no pressure monitoring of the interlid space during storage and no metal O-rings.

The storage cask consists of the following components (see Figures 1, 2, and 3):

- A containment vessel comprised of a cask body with shell, bottom, and primary and secondary lids.
- A pressure switch in the secondary lid with connection to the lid interspace (Version I only).
- Neutron shielding.
- Fuel basket assembly.
- Trunnions for handling.
- Penetrations in the bolted primary lid with sealed cover lids to facilitate draining, drying, backfilling, leak testing, and venting.
- Penetrations in the bolted secondary lid with sealed cover lids to facilitate leak testing and venting (Version I only).

Table 1. Design Basis Fuel Criteria

Depending on the cask load, the following criteria have to be satisfied:

<u>Cask Load without BPRAs in the Fuel Assemblies</u>

- Maximum 32 PWR fuel assemblies.
- The initial enrichment of each fuel assembly is ≤5.0 w/o U-235.
- Its burnup is no greater than 45 GWd/MTU.
- The minimum required cooling time is 2.9 years.
- Depending on the enrichment, some of the fuel assembly must be equipped with an absorber rod module before loading.

<u>Cask Load with BPRAs in the Fuel Assemblies (see Figure 28)</u>

- Maximum 32 PWR fuel assemblies.
- Depending on the enrichment, a maximum of 24 fuel assemblies can be provided with a BPRA; the remaining must have absorber rod modules installed before loading.
- The initial enrichment of the fuel assemblies defines the BPRA location in the basket.

- Closure bolts.
- Metallic O-ring seals on all bolted lids and penetration closures.

Through its design, the cask ensures structural integrity, subcriticality shielding, and retention of radioactive materials. The structural design will be in accordance with the ASME Code application. Decay heat is removed via the cask surface with fins to the environment by natural convection and radiation.

Dimensions and weights of the cask CASTOR® X/32 S are shown in Table 2. The materials used to fabricate the CASTOR® X/32 S cask are listed in Table 3.

Table 2. Dimensions and Weight of the CASTOR® X/32 S Cask

Dimension	mm	in
Overall Length	4,850	190.9
Outside Diameter	2,326	91.6
Cavity Diameter	1,730	68.1
Cavity Length	4,155	163.6
Body Wall Thickness	298	11.8
Primary Lid Thickness	260	10.2
Secondary Lid Thickness	80	3.16
Bottom Thickness	222	8.74
Bottom Plate Thickness	35	1.38
Moderator Rod Thickness	70	2.76
Cask Weight*:	**kg**	**lb**
Cask Weight (without Fuel)	81,922	180,557
Basket	14,539	32,052
Cask After Loading (without water and secondary lid)	104,322	229,926
Loaded On Storage Pad	106,774	235,330

*Weights determined theoretically, in practice deviation can occur.

3.1 Containment Vessel

The containment vessel for the CASTOR® X/32 S consists of a forged carbon-steel cylinder with an integrally-welded carbon-steel bottom as well as a bolted primary lid made of stainless steel with a metallic O-ring seal and a vent and drain opening closed by a lid with a metallic O-ring seal. A secondary lid is arranged over the primary lid. This secondary lid is different for Versions I and II and is described later in this paper.

The O-ring seating surfaces are overlaid with Inconel by welding to provide corrosion protection. These welded surfaces are machined to provide the necessary finish for the seating of the O-ring seals (see Figure 4).

The containment boundary components are shown in Figure 5. The overall containment vessel length is 4850 mm (190.6 inches) with a wall thickness of 298 mm (11.8 in). The cylindrical cask cavity has an internal diameter of 1,730 mm (68.1 inches) and an internal length of 4,155 mm (163.6 inches). The cask cavity is pressurized to 0.12 MPa (17.4 psi) with helium.

The secondary lid for Version I is made of stainless steel. It is fastened to the cask body by fifty-two bolts. The interspace between the primary lid and the secondary lid is used as part of the cask monitoring system (see Figure 6).

The secondary lid for Version II is made of forged carbon steel (like the cask body) and is attached to the cask body with a multipass narrow groove weld (see Figure 7a, 7b, 7c). There are no penetrations of the lid in Version II and the pressure in the interspace between the lids is not monitored.

3.2 Containment Monitoring (Version I only)

The interspace between the primary and the secondary lid is filled with helium and is used as part of the cask monitoring system. The initial pressure of the interspace is set to a pressure of 0.7 MPa (100 psi). A pressure switch is mounted in the secondary lid and is wired to an alarm system in the ISFSI.

3.3 Neutron Shielding

Neutron shielding on the cask is provided in both the radial and axial directions.

Neutron shielding in the radial direction is provided by

Table 3. Material Specification for the CASTOR® X/32 S

Item	Material	Specification
Cask body	Carbon steel	SA-508 4N, Class 3
Primary lid	Martensitic steel	SA-182, Grade F 6NM
Secondary lid (Version I)	Martensitic steel	SA-182, Grade F 6NM
Secondary lid (Version II)	Carbon steel	SA-508 4N, Class 3
Bottom plate	Stainless steel	SA-263 or 304
Lid bolts	Stainless steel	SA-479, Grade 414
Trunnion	Martensitic steel	SA-182, Grade F 6NM
Trunnion bolts	Stainless steel	SA-479, Grade 414
Fuel assembly receptacles	Stainless steel	SA-240 Grade 321
Heat transfer plates and absorber aod modules (ARMs)	Composite of Al/B4C	Minimum 0.09615 g/cc Boron 10
Gusset plates	Aluminum	SB-209
Basket restrainers	Martensitic steel	SA-182, Grade F 6NM
Bolts for structural support	Stainless steel	SA-479, Grade 414
Metallic O-ring	Al/Inconel Helicoflex®	
Neutron shielding	Polyethylene	WS 1-PE-HD-01, Ind. 02
Cask cavity inside coating	Aluminum flame spray	Report No. 2000321
Cask outside coating	BROCOPHAN	Brocolor® Lackfabrik

polyethylene rods set into two concentric rows of axial bore holes of 70 mm (2.75 inches) diameter in the wall of the cask body. Each concentric row contains sixty-two boreholes for a total of 124 boreholes. (see Figure 2). The boreholes in the two concentric rows are offset to provide an unbroken line of neutron shielding for radiation from the cask cavity.

Neutron shielding in the axial direction at the lid and bottom area is provided by polyethylene disks.

3.4 Fuel Basket

The fuel basket provides support of the fuel assemblies, control of criticality, and a path to conduct heat from the fuel assembly to the cask body.

The configuration of the fuel basket in the cask cavity is illustrated in Figure 8. The fuel basket is designed to accommodate up to thirty-two intact PWR fuel assemblies. Fuel receptacles are manufactured by the welding of stainless steel plates to enclose and secure the fuel assemblies. The stainless steel fuel receptacles are held in place by a basket gridwork of borated aluminum plates and gusset plates assembled inside the cask cavity (see Figure 9).

The borated aluminum plates of this basket gridwork provide heat conductivity. The boron content of these plates assures safety of nuclear criticality.

Except for the welding along the length of the stainless steel fuel receptacles, the fuel basket for the CASTOR® X/32 S is manufactured with no weldings. The fuel receptacles and borated aluminum plates are manufactured to close tolerances. Mechanical techniques are used to form the basket gridwork and to assemble it together with the fuel receptacles inside the body cavity of the CASTOR® X/32 S.

GNB has already fabricated a prototype basket to show the feasibility of this basket concept (see Figures 10, 12, and 13).

The closely fitting borated aluminum plates of the basket gridwork fix the square-shaped stainless steel fuel receptacles in a central position inside the cask cavity. This compact close-tolerance arrangement is intended to minimize the movement of the fuel assemblies relative to each other and to the cask body under normal, off-normal, and accident conditions.

Stainless steel supports (basket restrainer) are bolted to the top of the cask cavity to protect the primary lid from dynamic loads coming from the weight of the gusset plates and the outer row of fuel assembly receptacles (see Figure 3).

Table 4. Component Temperatures for Conditions of Normal Storage with an Initial Decay Heat Load of 32 kW

Components	Maximum(°C)
Cask Wall (ID)	114
Cask Wall (OD)	98
Bottom Plate	119
Primary Lid Seal	80
Secondary Lid Seal	75
Fuel Cladding	258
Fuel Basket	239
Aluminum Gusset	121
Inner Moderator Rods	112
Outer Moderator Rods	105

3.5 Operational Features

The CASTOR® X/32 S cask is designed for dry storage of thirty-two intact PWR fuel assemblies and six associated BPRAs.

Each fuel assembly is assumed to have a maximum initial enrichment of 5 percent by weight of U-235. Further, the fuel is assumed to have a maximum of 45,000 MWD/MTU burnup, a minimum storage time of between 2.9 and 9.5 years in the spent fuel pool after reactor discharge, and a maximum decay heat load of 1.0 kW per assembly. If the assembly is loaded together with a BPRA, the total heat load of the assembly and BPRA must not exceed 1 kW. Thus, the CASTOR® X/32 S has a total heat dissipation capability of 32 kW. The heat rejection dissipation capability of the cask CASTOR® X/32 S maintains the maximum fuel rod cladding temperature below the limit of 342°C. The calculated maximum fuel rod cladding temperature of 258°C is based on normal operating conditions with a 32 kW decay heat load, 27°C (80°F) ambient air and insolation. The fuel assemblies are stored in an inert helium gas atmosphere.

The shielding features of the cask CASTOR® X/32 S are designed to maintain the average surface combined gamma and neutron dose rate to less than 2 mSv/h (200 mrem/hr) under normal operating conditions.

The criticality control features of the cask CASTOR® X/32 S are designed to maintain the neutron multiplication factor k-effective (including uncertainties and calculational bias) at less than 0.95 under normal, off-normal, and accident conditions.

Depending on the enrichment, some of the fuel assembly must be equipped with specifically engineered insertable ARMs for additional nuclear criticality safety.

4. Result of the Thermal Analysis

Thermal Evaluation for Normal Storage Conditions

The CASTOR®X/32 S was examined under a maximum internal heat load of 32 kW using the three-dimensional ANSYS-FEA model (see Figures 13, 14, 15, and 16). The temperatures of fuel assemblies and cask components were found to be within the specified limits. The temperature contours are shown in Figures 17, 18, 19, and 20.

The highest temperature in the model occurs in the inner fuel assembly and has a value of about 258°C (496°F). Figure 21 shows a plot of the radial temperature distribution at the highest temperature cross-section.

Figure 22 shows a plot of the thermal flux at the highest temperature cross-section. Table 4 is a tabulation of the maximum temperatures for components in the CASTOR, X/32 S.

Thermal Evaluation for Accident Conditions

Fire Event

Figure 23 shows the temperature history of the cask components during the fire event.

Burial Event

Figure 24 shows the temperature history of the cask components during the burial event. In the burial event, the cask cannot dissipate heat to its surroundings and will, therefore, continue to heat up until it is uncovered. The FEA results demonstrate that the maximum operating temperature specified for the lid seal of the Version I cask of 300°C (572°F) is not exceeded up to 100 hours after burial. Similarly from Figure 26 it is seen that the fuel temperature does not exceed the limit of 570°C (1,058°F) up to 100 hours after burial.

Figure 25 shows the temperature history of the cask components during the 100 hour analyzed period of the loading event.

5. Results of the Shielding Analysis

Shielding for the CASTOR® X/32 S is provided by the thick-walled cask body and the lid system. For neutron shielding, polyethylene rods are arranged in longitudinal boreholes in the vessel wall and polyethylene-plates are inserted between the primary and secondary lid and between the cask bottom and bottom steel plate. Additional shielding is provided by the basket structures.

The source terms for the design spent PWR fuel and sixteen assumed BPRAs are determined using ORIGEN-2.1. The shielding analyses are performed with MCNP-4B and the model shown in Figures 26 and 27.

The expected maximum dose rates from the CASTOR® X/32 S for normal and off-normal conditions are provided in Table 5.

6. Results of the Criticality Analysis

It is demonstrated by KENO Va-Code that the effective neutron multiplication factor (k_{eff}) of the CASTOR® X/32 S, including all biases and uncertainties does not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Criticality safety of the CASTOR® X/32 S depends on the following principal design criteria (see Table 6):

- The geometrical structure of the fuel basket.
- The permanent fixed neutron-absorbing materials in the fuel basket structural parts.
- A limitation of the maximum enrichment for PWR Uranium fuel.
- The neutron absorber rod module inserted in some guide tubes of the fuel assembly.
- Soluble boron during cask loading to cover possible misloadings.
- The neutron-absorbing material is used in both the fuel basket structural parts and the ARMs inserted in the guide tubes of the intact PWR Zircaloy-clad fuel assemblies. For this material, a minimum boron-10 content of 0.09615 gm/cc (prior to reduction in the analytical models) is required. The analysis utilizes only 75 percent of this as recommended in NUREG-1536.

The CASTOR® X/32 S is designed for dry storage (no moderating water present in the cavity), and under storage conditions the reactivity is very low (k_{eff} < 0.55). However, the CASTOR® X/32 S is also designed for later transport, and thus, the flooded state is the limiting case in terms of reactivity.

Optimum moderation (fresh water with a density of 1 g/cm^3) is considered in performing the criticality analyses. However, since the cask will later be licensed for transport, nonuniform flooding due to a tilted cask in a body of fresh water will be assumed.

With regard to the fresh fuel, credit is taken for U-234 but no credit is taken for the U-236 content. The fuel stack density is assumed to be 96 percent (10.5216 g/cm^3) of theoretical for all criticality analyses. No credit is taken for fuel pellet dishing or chamfering.

The fuel basket with its thirty-two fuel assembly positions is designed such that the neutron-absorbing material is fixed and will remain effective for storage periods greater than twenty years. There are no credible conditions that will displace the neutron-absorbing material.

Criticality safety of the CASTOR® X/32 S does not rely on the use of:

- fuel-related burnable neutron absorbers; or
- more than 75 percent of the B-10 content for the fixed

Table 5. Normal/Off-Normal Surface Dose Rates

Dose Rates at Cask Surface for Normal and Off-Normal Conditions Cask Content: 32 Design Basis Fuel Assemblies Plus 16 Inserted Design Basis BPRAs						
Dose point location	**Gamma dose rate**		**Neutron dose rate**		**Total dose rate**	
	μSv/h	**(mrem/h)**	**μSv/h**	**(mrem/h)**	**μSv/h**	**(mrem/h)**
Lid; center	210	(21)	130	(13)	340	(34)
S_{top}; averaged	1,290	(129)	370	(37)	1,660	(166)
S_{mid}; averaged	1,350	(135)	600	(60)	1,950	(195)
S_{bot}; averaged	1,250	(125)	160	(16)	1,410	(141)

Table 6. Criticality Results

Number of ARMs	Enrichment Limit (wt % U-235)	Fuel Manufacture and Type	Calculated k_{eff}	Uncertainty in Calculated k_{eff} (1σ)	Value to Compare to USL of 0.9413
CE 14X14 Plants					
8	3.01	CE – Standard	0.9365	0.0007	0.9379
		Westinghouse	0.9402	0.0004	0.9410
		ANF	0.9319	0.0012	0.9343
16	3.60	CE – Standard	0.9352	0.0012	0.9376
		Westinghouse	0.9388	0.0004	0.9396
		ANF	0.9329	0.0012	0.9353
20	4.06	CE – Standard	0.9356	0.0004	0.9364
		Westinghouse	0.9392	0.0004	0.9400
		ANF	0.9325	0.0004	0.9333
24	4.33	CE – Standard	0.9353	0.0014	0.9381
		Westinghouse	0.9372	0.0005	0.9382
		ANF	0.9298	0.0011	0.9320
28	5.00	CE – Standard	0.9342	0.0014	0.9370
		Westinghouse	0.9367	0.0008	0.9383
		ANF	0.9289	0.0012	0.9313
St. Lucie Unit 2 (CE 16X16 short)					
8	3.00	CE	0.9402	0.0004	0.9410
16	3.51	CE	0.9402	0.0004	0.9410
20	3.94	CE	0.9400	0.0004	0.9408
24	4.20	CE	0.9399	0.0004	0.9407
28	4.74	CE	0.9370	0.0004	0.9378
32	5.0	CE	0.8953	0.0008	0.8969
Westinghouse 14X14 Plants					
8	3.36	W – Standard	0.9356	0.0009	0.9374
		W – OFA	0.9396	0.0004	0.9404
		ANF – Standard	0.9296	0.0012	0.9320
		ANF Top Rod	0.9299	0.0011	0.9321
16	4.22	W – Standard	0.9345	0.0008	0.9361
		W – OFA	0.9388	0.0007	0.9402
		ANF – Standard	0.9267	0.0012	0.9291
		ANF Top Rod	0.9290	0.0013	0.9316
20	4.83	W – Standard	0.9335	0.0009	0.9353
		W – OFA	0.9392	0.0008	0.9408
		ANF – Standard	0.9259	0.0013	0.9285
		ANF Top Rod	0.9265	0.0013	0.9291
28	5.00	W – Standard	0.9260	0.0013	0.9286
		W – OFA	0.9305	0.0013	0.9331
		ANF – Standard	0.9179	0.0013	0.9205
		ANF Top Rod	0.9180	0.0013	0.9206

neutron absorber in the cask components.

The worst hypothetical combination of dimensional tolerances (most conservative values within the range of acceptable values) is considered for analyses, in compliance to NUREG-1536.

Criticality analyses assume the use of ARMs located in guide tube positions of assemblies in specific locations in the cask.

Maximum neutron multiplication factors k_{eff} (max) resulting from criticality analyses considering CASTOR®X/32 S cask loads consisting of thirty-two PWR fuel assemblies and varying number of ARMs are presented in Table 6. The ARM arrangements are found in Figure 28.

7. Results of the Structural Analysis

The stresses in all structural components of the CASTOR® X/32 S due to the loads of normal operations and accident conditions have been determined by a combination of closed form calculations and finite element analysis. Other considerations such as basket deflection (for criticality purposes) and fracture toughness for the cask body and the weldings and fuel rod creep and stresses are also analyzed.

The results of the analyses of the stresses induced in components of the CASTOR® X/32 S under the loads of normal operations and accident conditions are tabulated as an example in Table 7 together with the ASME Code allowable stresses for comparison.

The structural analyses demonstrate that the CASTOR® X/32 S meets all requirements of the ASME Code under the loads of normal service and accident conditions. Figures 29 through 32 show the ANSYS model and the stress intensity contour plots. Figure 33 shows the tangential creep of the cladding material versus storage time.

The trunnions are capable of meeting the criteria of ANSI 14.6. All loading conditions are conservative and envelop the actual and hypothetical conditions specified for this cask.

Figures 34 and 35 show the FEM trunnion model and the stress intensity plott.

8. Conclusion

The results of the thermal, shielding, criticality, and structural analysis have shown that the cask design is in full compliance with all applicable federal requirements.

9. Further Activities of GNB in the Field of Storage and Transportation of Spent Fuel

Further cask development and licensing steps are shown in the time schedule (see Figure 36).

The storage license for the PWR cask will be available in 2002.

The SAR for the the transportation cask will be prepared in this time and will be applied to the NRC in first quarter of 2001. For this application the intention of GNB is to use burnup credit for this cask. In case we use burnup credit, we can remove most of the ARMs out of the fuel assemblies; this gives more space for additional BPRAs. The transport license for this cask will be available in early 2003.

GNB intends to cooperate with U.S. fabricators to start the production of the prototype cask after receiving a positive evaluation from the NRC for the Transport SAR in early 2002.

The licences for the BWR-cask will be available after issuing the PWR-cask license.

Table 7. Summary of Maximum Stress Intensity — Case NC1 and NCT1 CASTOR® X/32 S — Cask Body and Lids

Service Condition	Component	Stress Category	Maximum Stress Intensity MPa (ksi)	Allowable Stress Limits MPa (ksi)	Margin to Allowable
Normal Mechanical Loads Only	Cask body	P_m		206.9 (30.0)	5.3
		$P_m + P_b$	39.3 (5.7)	310.3 (45.0)	7.9
	Primary lid	P_m		264 (38.3)	38.8
		$P_m + P_b$	6.8 (1.0)	396 (57.5)	58.2
	Secondary lid (Version I)	P_m		264 (38.3)	3.2
		$P_m + P_b$	81.9 (11.9)	396 (57.5)	4.8
	Secondary lid (Version II)	P_m		206.9 (30.0)	2.5
		$P_m + P_b$	81.9 (11.9)	310.3 (45.0)	3.8
Normal Mechanical Plus Thermal	Cask body	$P_m + P_b + Q$	60.5 (8.7)	620.7 (90.0)	10.2
	Primary lid	$P_m + P_b + Q$	22.9 (3.3)	792 (115.0)	34.6
	Secondary lid (Version I)	$P_m + P_b + Q$	87.3 (12.7)	792 (115.0)	9.1
	Secondary lid (Version II)	$P_m + P_b + Q$	87.3 (12.7)	620.7 (90.0)	7.1

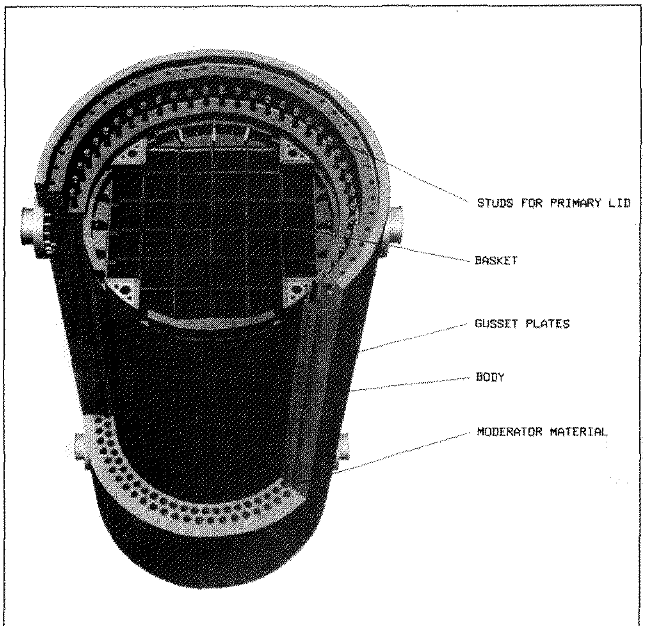

Figure 1. View into the Cask Cavity

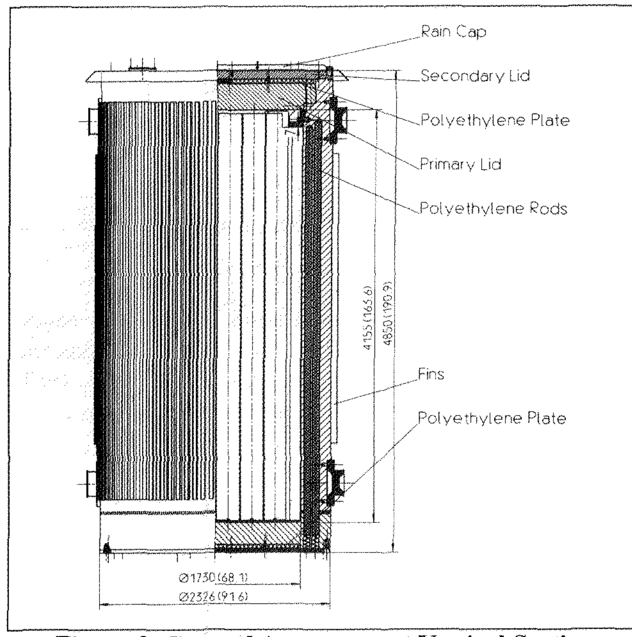

Figure 2. General Arrangement Vertical Section

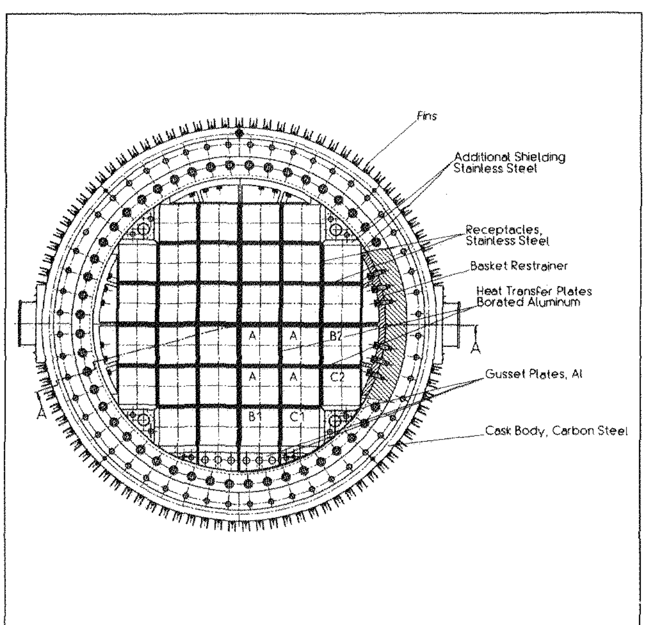

Figure 3. General Arrangement Cross-section

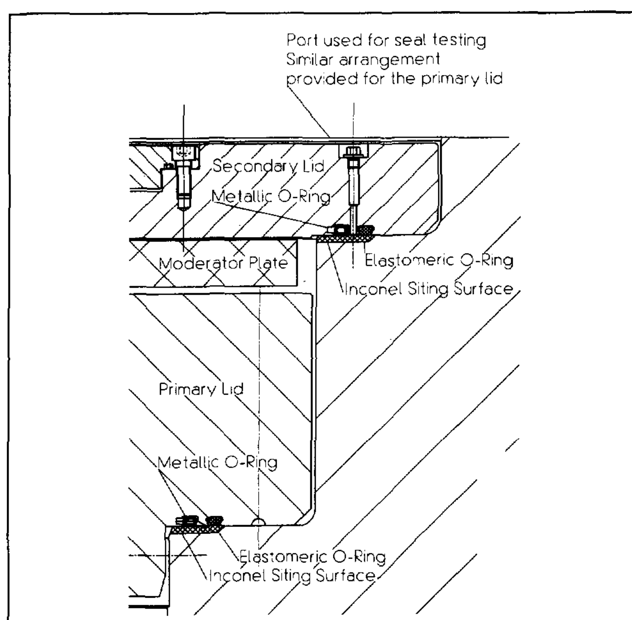

Figure 4. Seal Arrangement for Primary and Secondary Lid

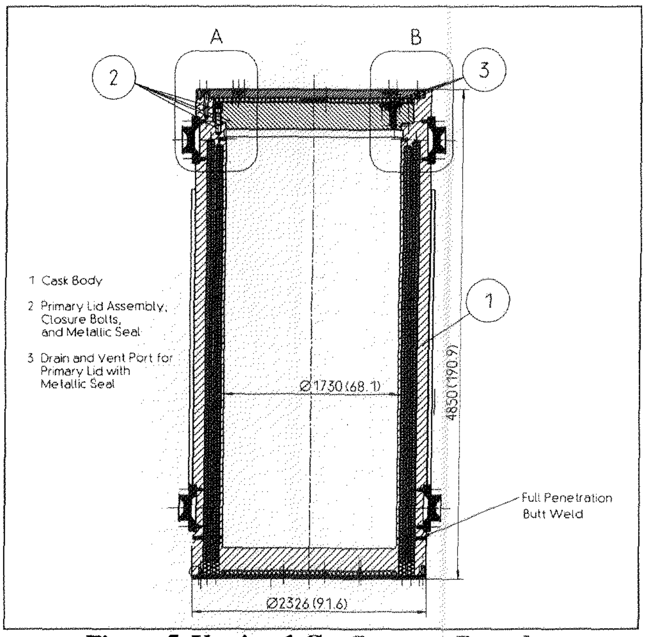

Figure 5. Version 1 Confinement Boundary

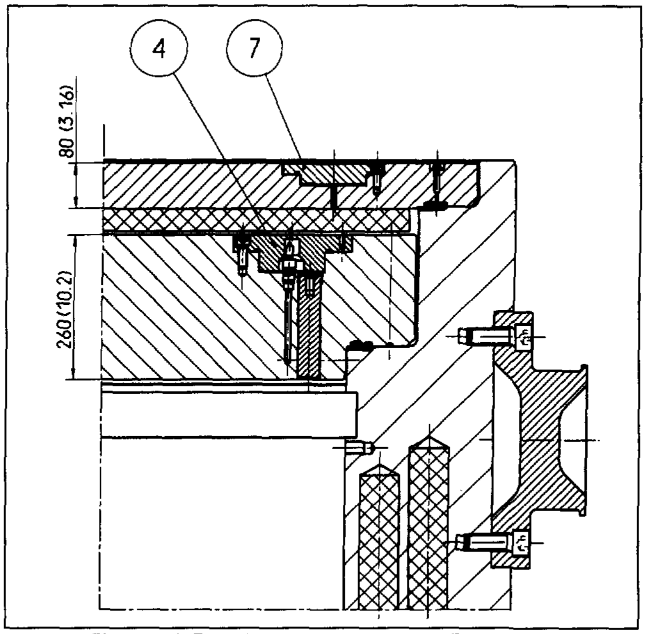

Figure 6. Lid Assembly, Version I Detail B

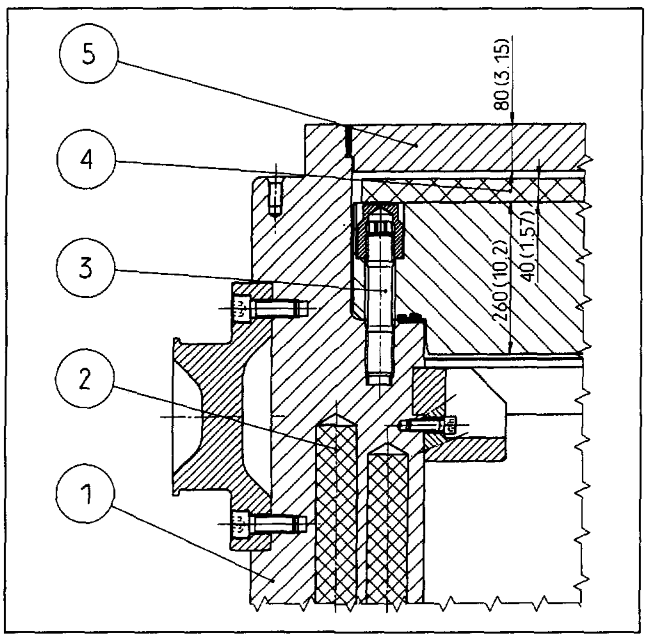

Figure 7a. Lid Assembly, Version II Welded Closure (Secondary Lid)

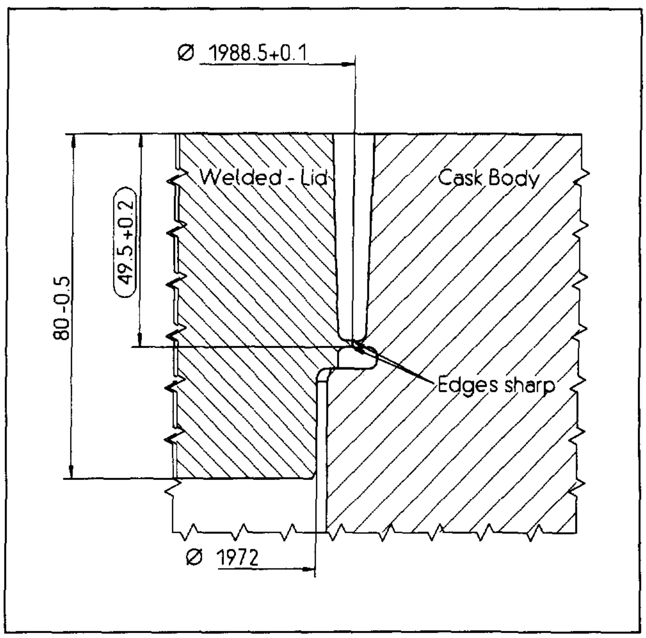

Figure 7b. Detail Drawing Secondary Lid, Version II

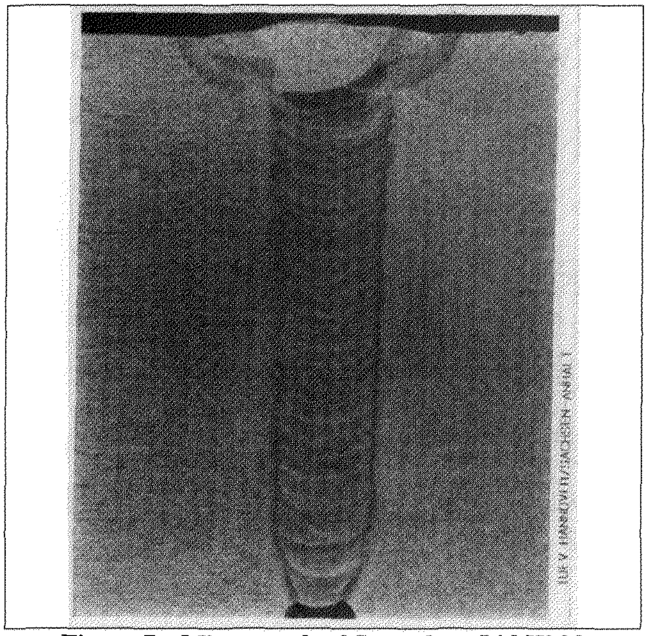

Figure 7c. Micrograph of Secondary Lid Weld

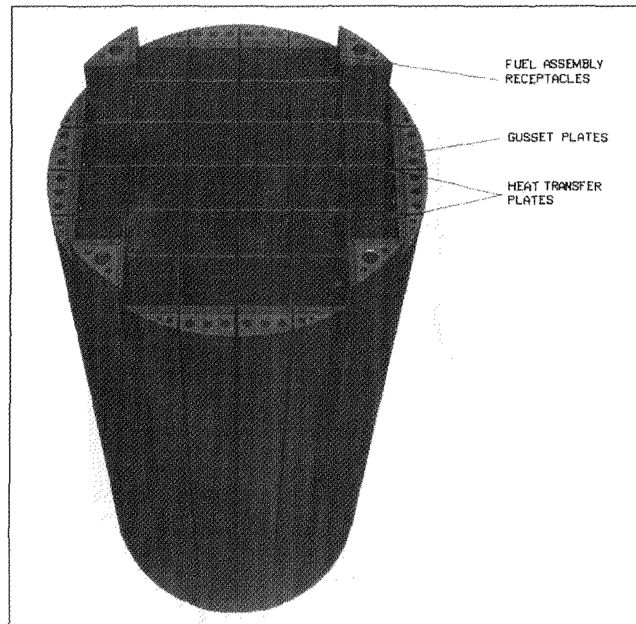

Figure 8. Fuel Assembly Basket

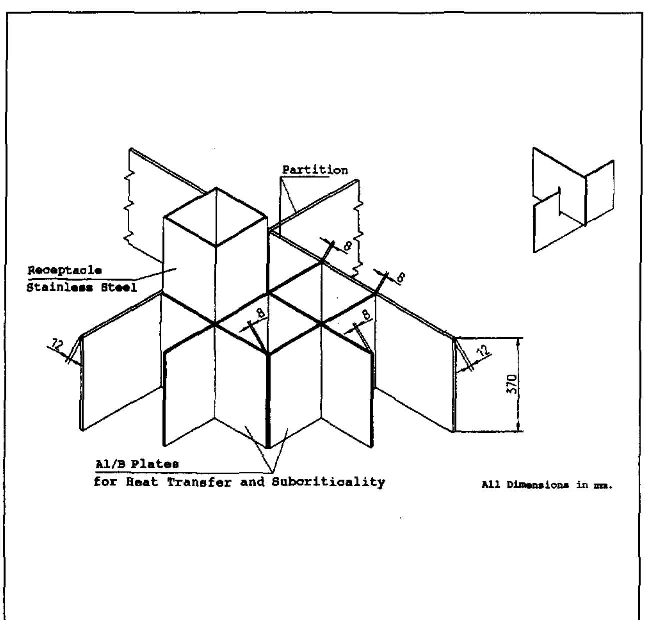

Figure 9. Partial Basket Layer General Arrangement

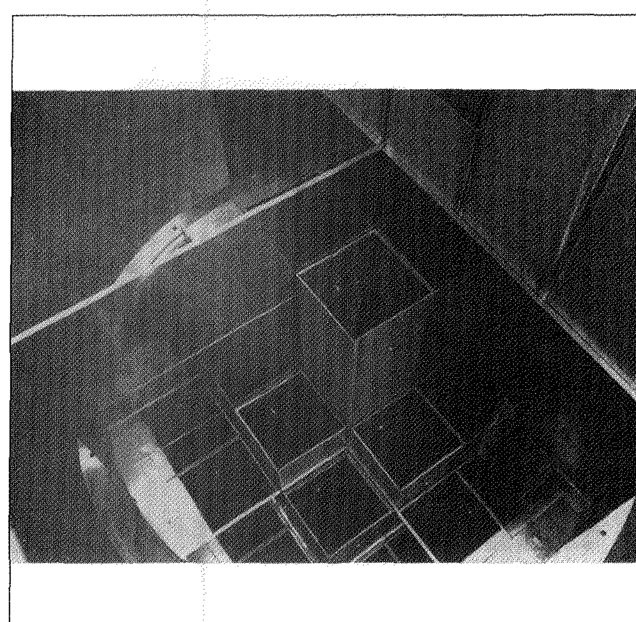

Figure 10. Basket Side View

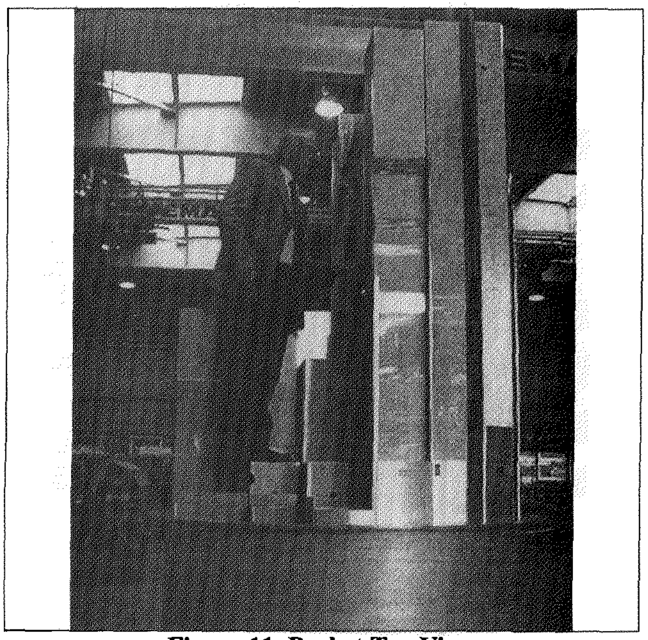

Figure 11. Basket Top View

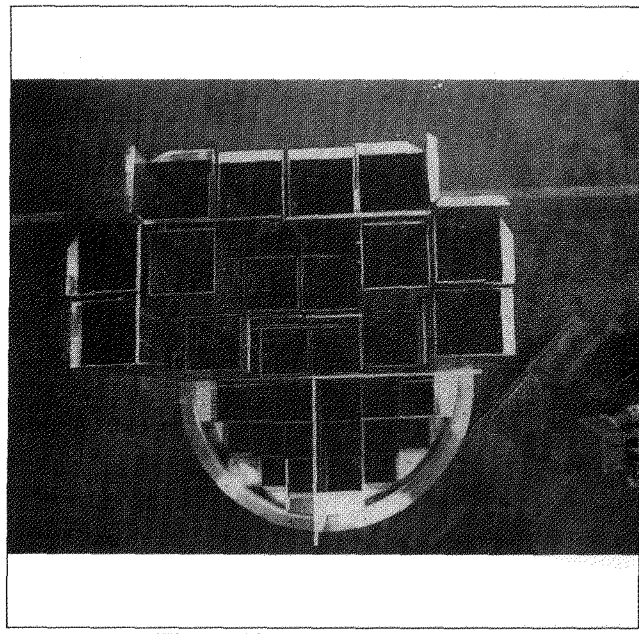

Figure 12. Basket Detail View

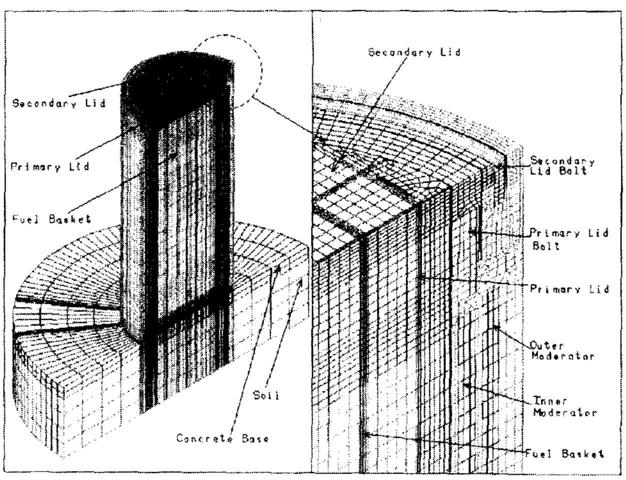

Figure 13. Overall Finite Element Model

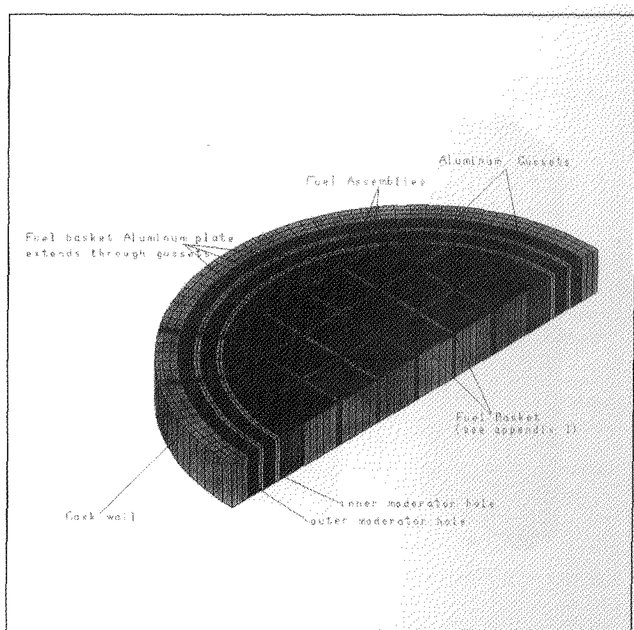

Figure 14. Detail of Typical Cross-section

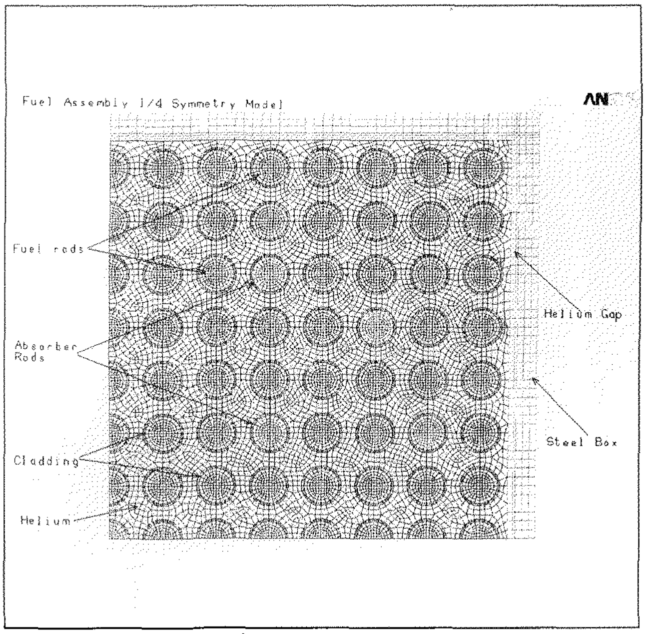

Figure 15. Cask Wall Model with Moderator Holes

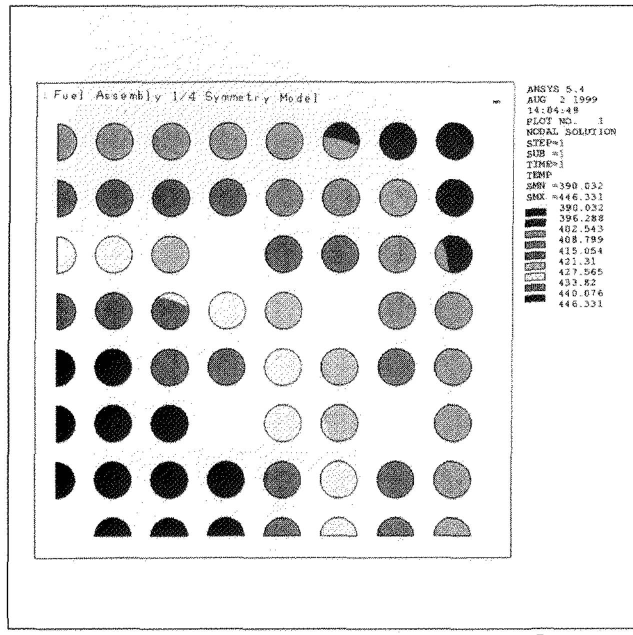

Figure 16. Fuel Assembly 1/4 Symmetry Model

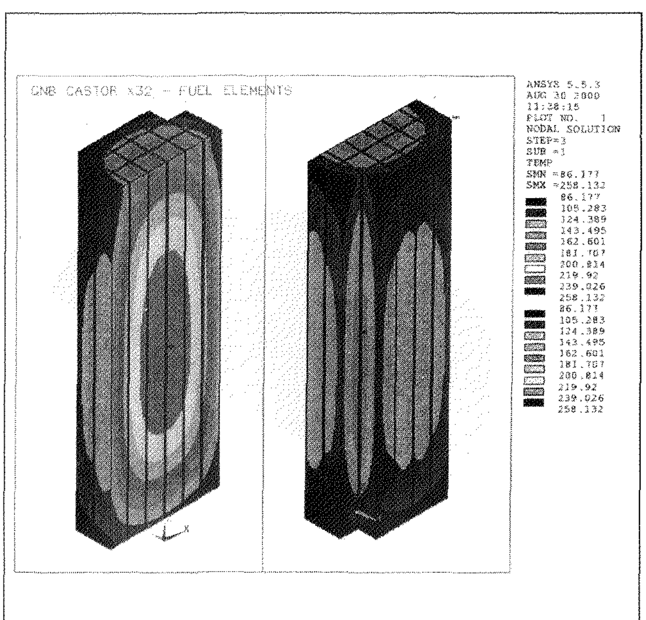

Figure 17. Fuel Rod Temperature

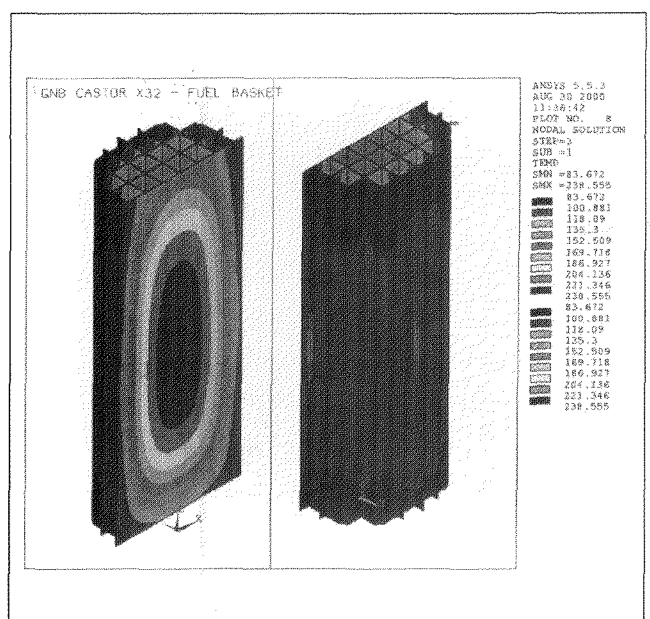

Figure 18. Normal Storage — Temperatures for Fuel Assembly

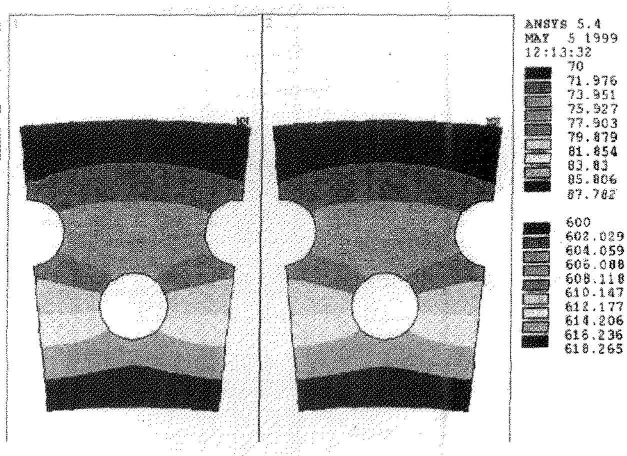

Figure 20. Temperature Contours for Models with Holes

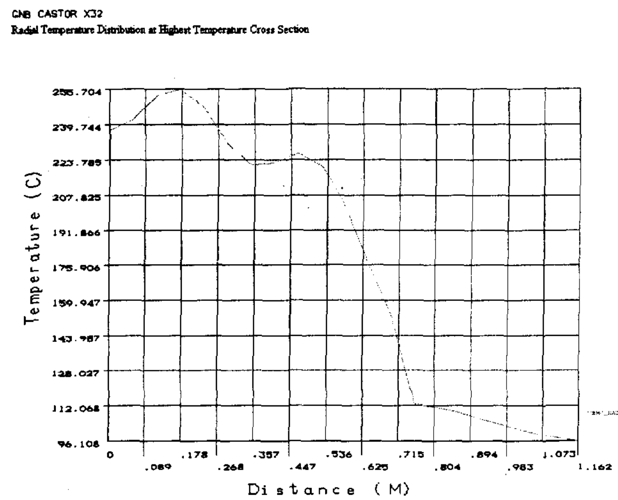

Figure 21. Normal Storage — Radial Temperature Distribution at Highest-Temperature Cross-section

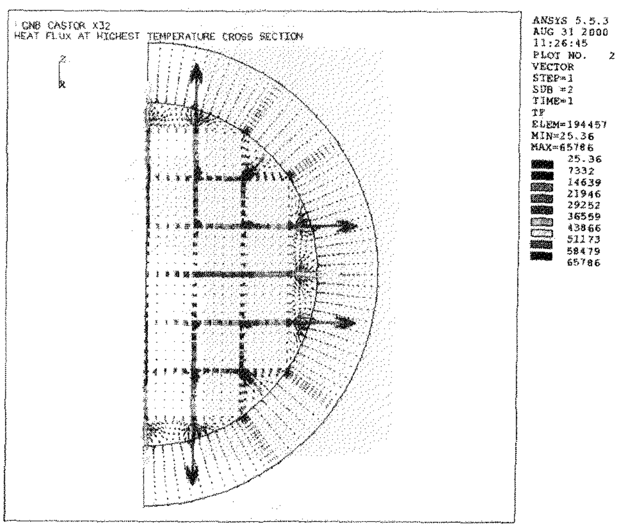

Figure 22. Normal Storage — Heat Flux Plot at Highest-Temperature Cros-section

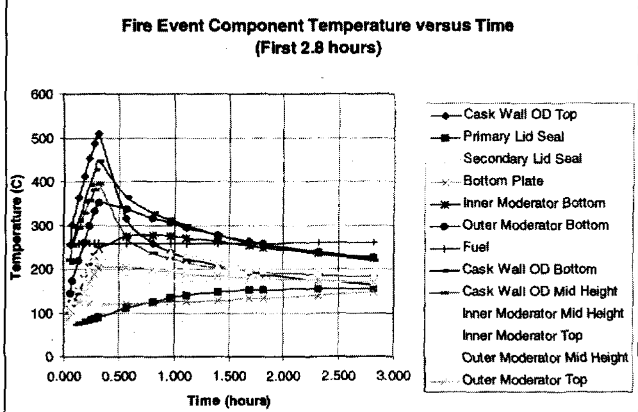

Figure 23. Fire Event, Component Temperature versus Time

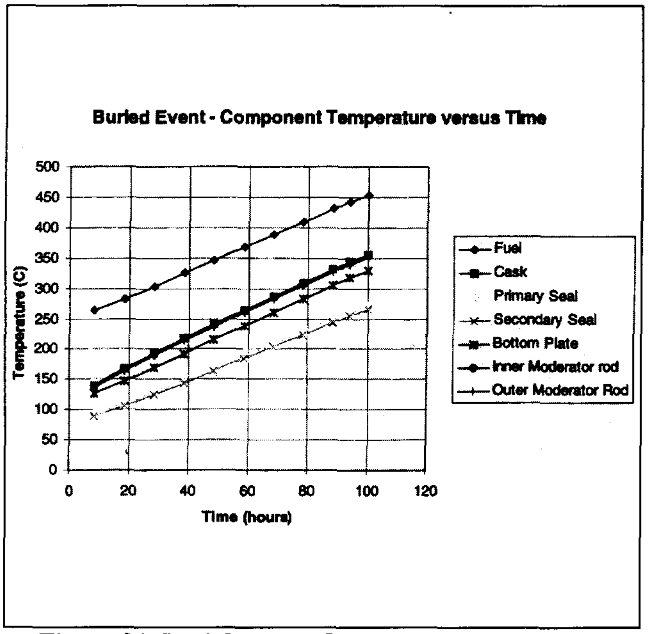

Figure 24. Burial Event, Component Temperature versus Time

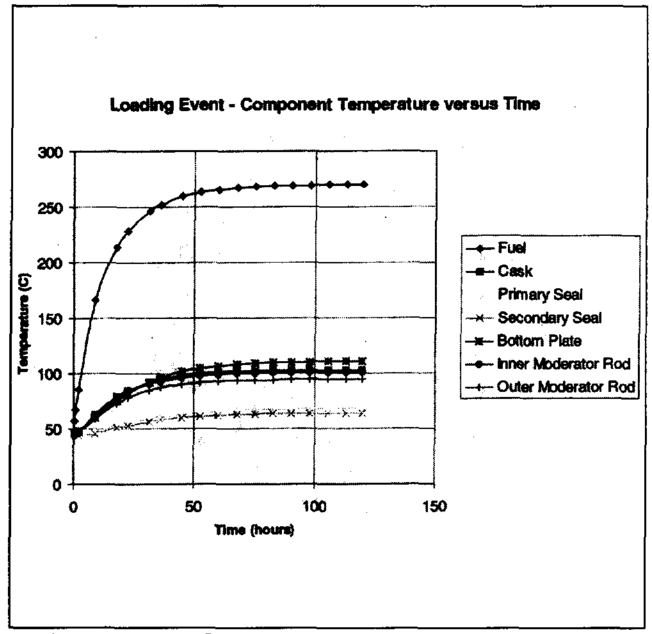

Figure 25. Loading Event, Component Temperature versus Time

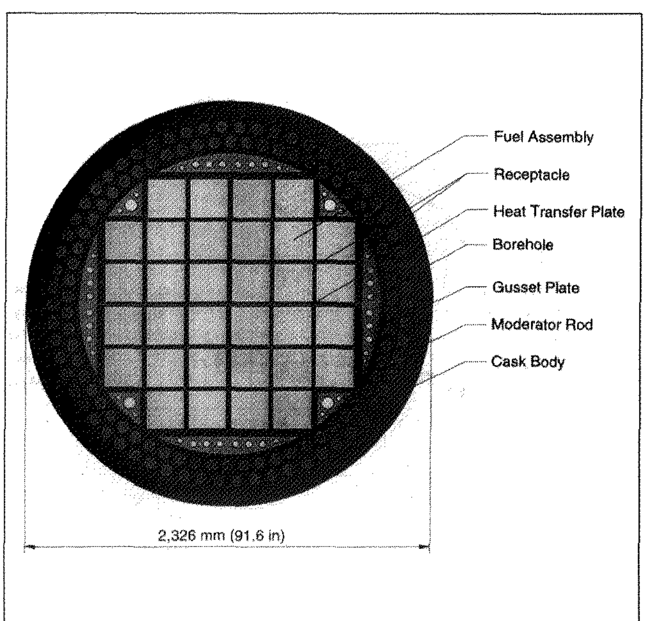

Figure 26. Radial Cross-section of the MCNP-B-Model

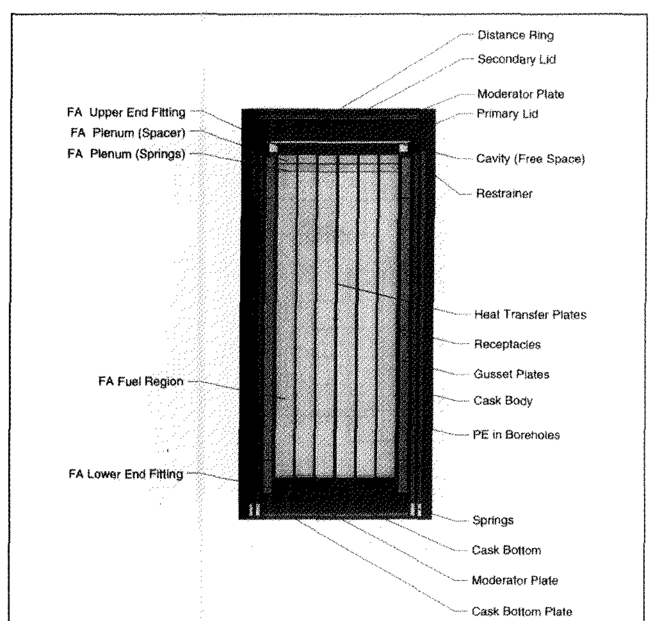

Figure 27. Axial Cross-section of the MCNP-4B-Model

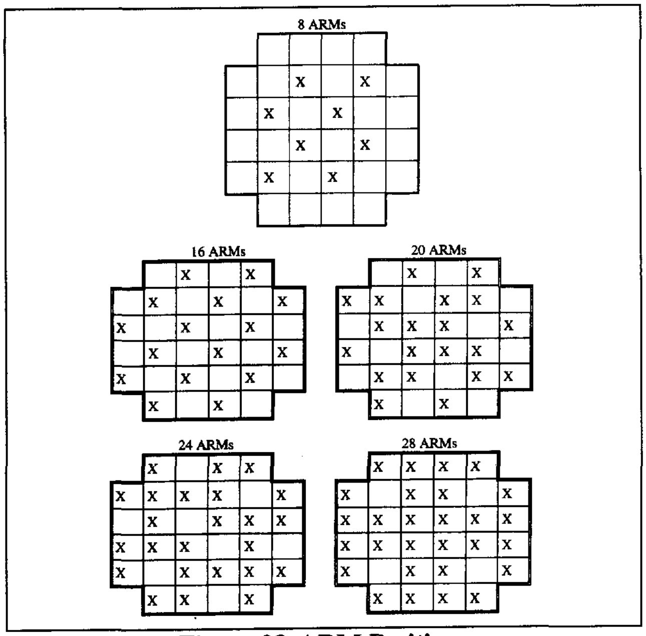

Figure 28. ARM Positions

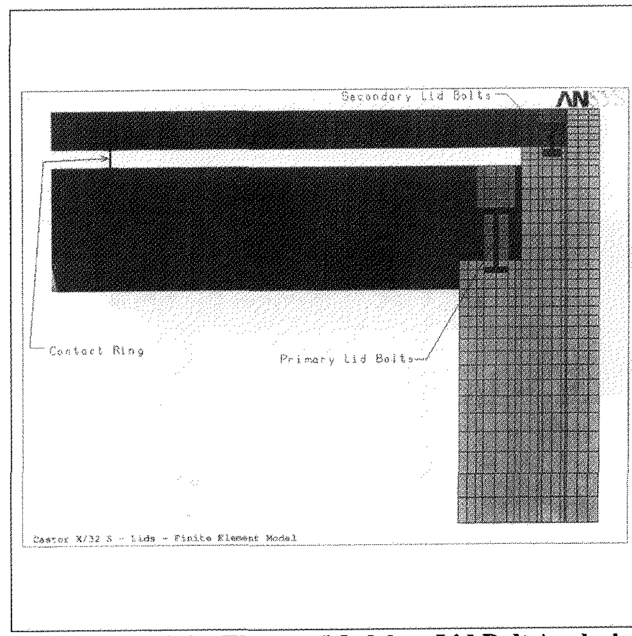

Figure 29. Finite Element Model — Lid Bolt Analysis

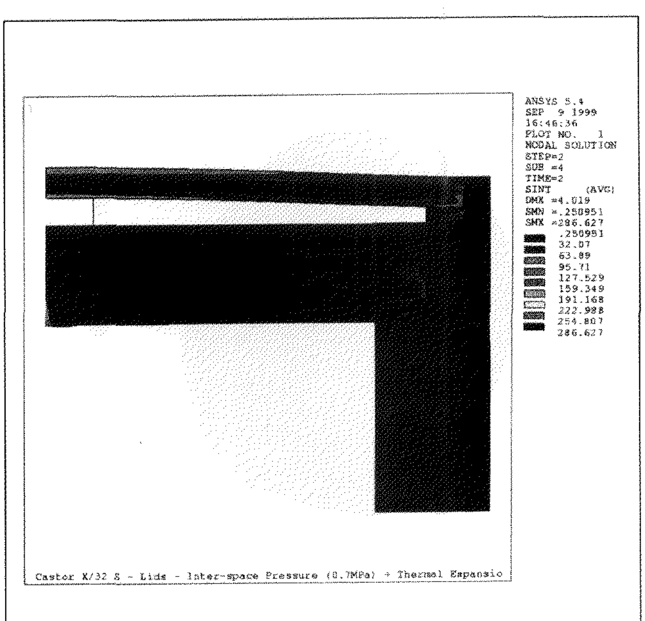

Figure 30. Stress Intensity Contour Plot — Interspace Pressure of 0.7 MPa

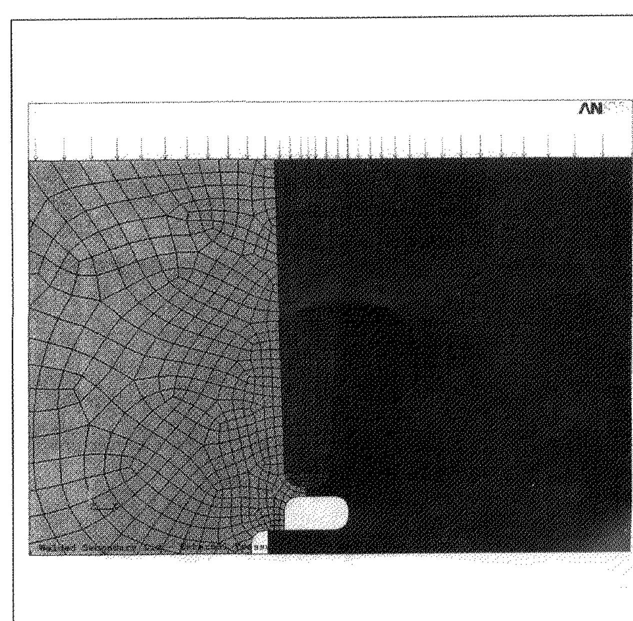

Figure 31. Finite Element Model — External Pressure 2.0 MPa

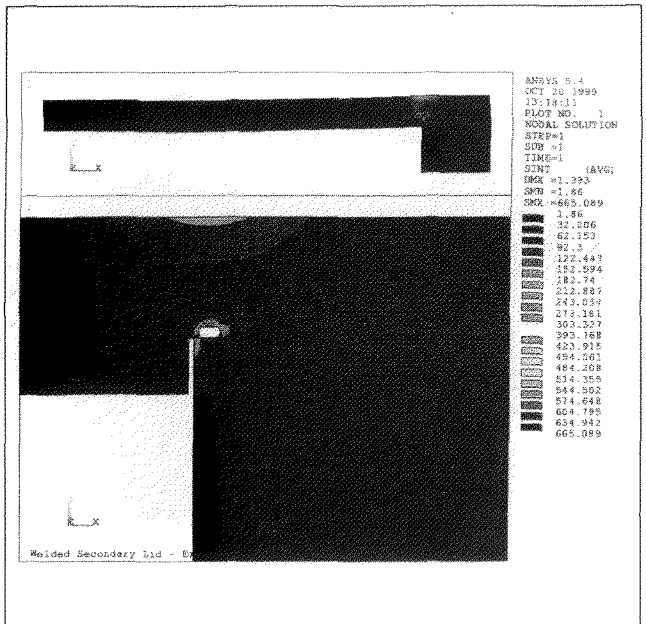

Figure 32. Welded Secondary Lid — External Pressure — Stress Intensity

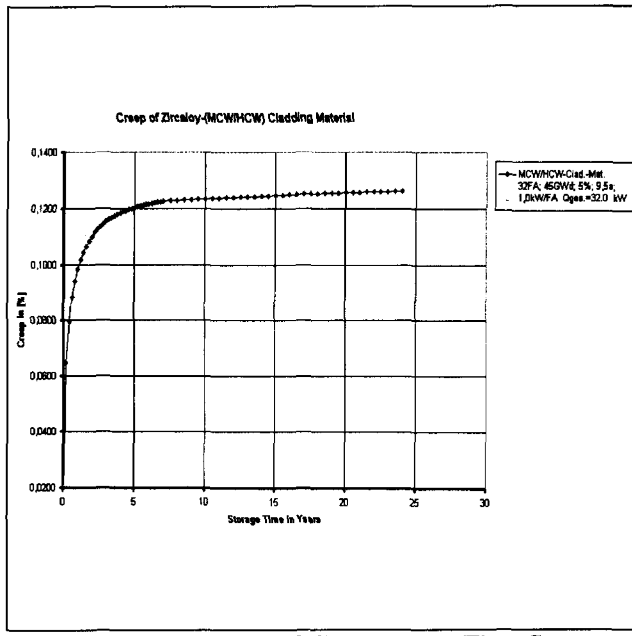

Figure 33. Tangential Creep versus Time Start Temperature 227°C

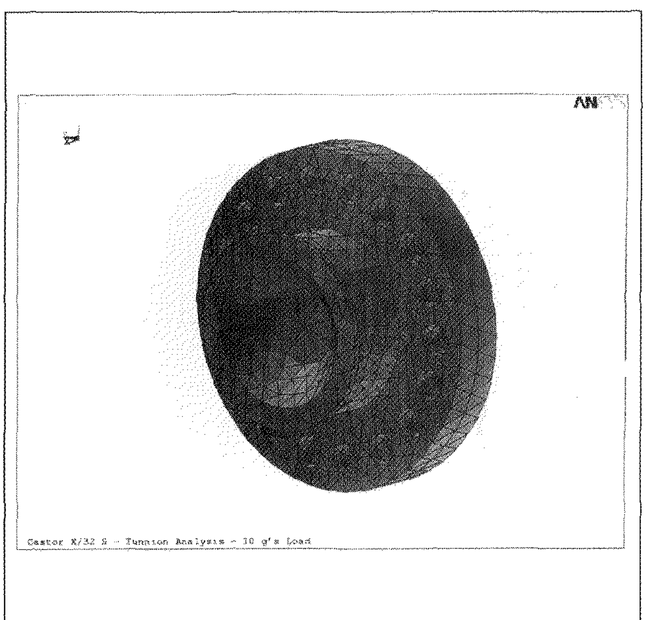

Figure 34. Finite Element Plot of Trunnion with 10g Loading

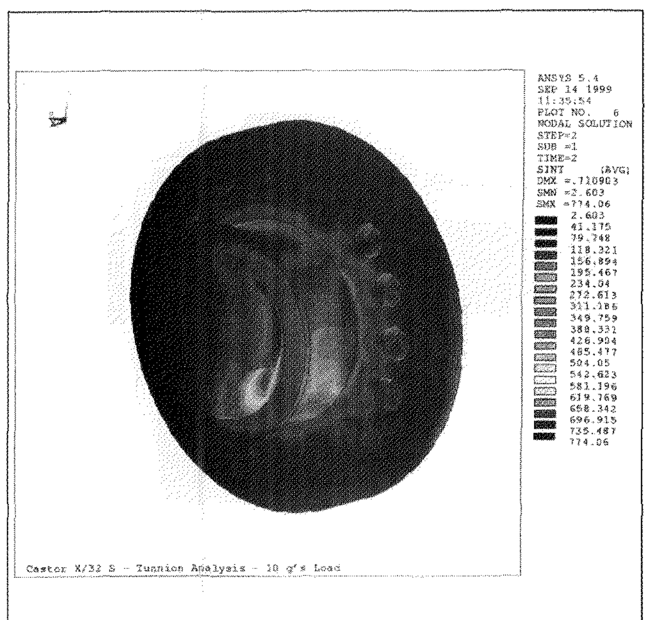

Figure 35. Stress Intensity (MPA) Plot Due to the 10g Lifting Load (Front View)

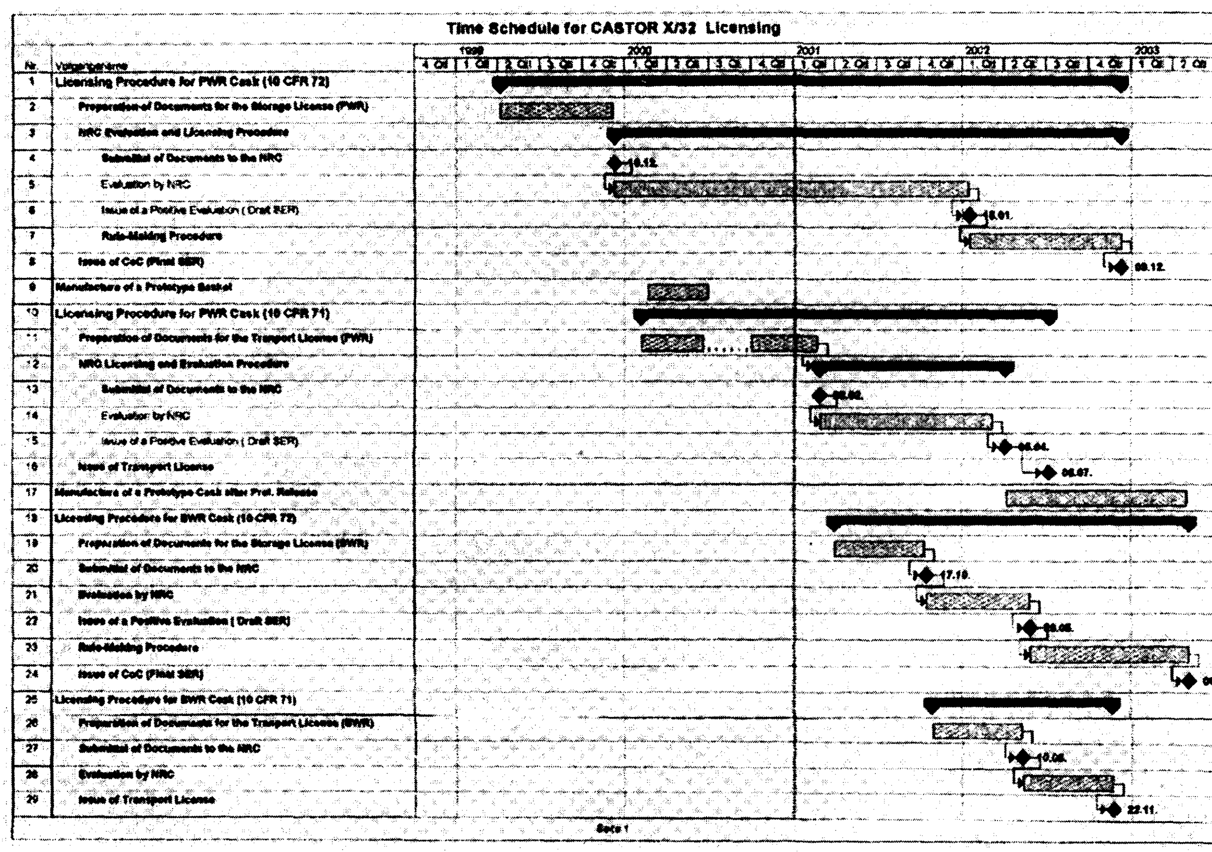

Figure 36. Time Schedule

Spent Fuel Storage Overviews on a Worldwide Basis

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Vienna, Austria

1. Introduction

Spent nuclear fuel is indeed in the very center of the discussions about energy, environment, and security.

In this context, the IAEA has played an important role since it was created upon the initiative of President Eisenhower as a response to his call for Atoms for Peace.

The IAEA has, at present, 130 member states that share the interest in the peaceful use of the atom. However, related to nuclear energy and the fuel cycle, their interests may differ considerably. It is evident that with regard to the peaceful use of the atom, the member states consider the issue of spent-nuclear fuel very important. There are not only the countries with installed nuclear power programs that have an interest in nuclear fuel issues, but also those considering embarking on nuclear power. There are those that do not have nuclear power but are located along or close to transport routes of spent fuel and those that have potential or actual fuel-cycle facilities close to their borders.

In the first part, the spent fuel arising (globally and per region) is assessed and put in balance with the storage capacities available, in the second part, some information is given on the status of AFR-storage facilities in different countries.

2. Status of Spent Fuel Arising

The data and forecasting on spent fuel arising are based on IAEA statistics and forecasts on the power generation, the amounts of spent fuel actually generated, and forecasts of the amount to be generated.

2.1. Status of Nuclear Power

Today the growth of nuclear power is at a standstill in Western Europe and North America, while it is expanding in parts of Asia and Eastern Europe. At the end of last year, 433 nuclear reactors were operating in thirty-one countries. They provided about 17 percent of the global electricity supply.

The total operating nuclear capacity by the end of 1999 was 349 GWe, thirty-seven reactors were under construction with a total additional capacity of 31 GWe.

2.2. Spent fuel arising at the global scale

The total amount of spent fuel cumulatively generated worldwide at the end of 2000 was expected to be close to 230,000 metric tons of heavy metal, (tHM). The annual discharge rate is about 10,500 tHM.

Projections indicate that the cumulative amount generated by the year 2010 may be close to 340,000 tHM. (See Figure 1.)

By the year 2020, the period when most of the presently operated nuclear power reactors will be close to the end of their designed operation life time, the total quantity of spent fuel generated will be approximately 445,000 tHM.

In the assumption that, for the near future, the used global reprocessing capacity will be in the order of 4,000 metric tons per year, we can state that the quantity of the stored spent fuel at present is about thirty-five times the reprocessing capacity. In other words, we would need some thirty-five years to reprocess all the spent fuel that has been stored to date.

2.3 Spent-fuel storage in world regions

On a regional basis, the picture for the spent fuel stored and to be stored is as follows. (See Figure 2.)

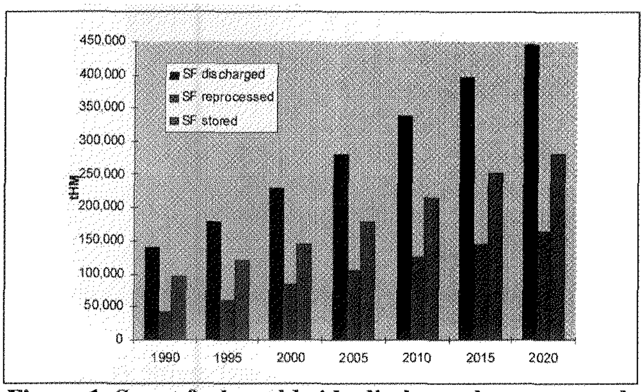

Figure 1. Spent fuel worldwide discharged, reprocessed, and stored

By the end of 2000 the amount of the spent fuel that we expect to be in storage is:

Western Europe: about 32.5 ktHM

Eastern Europe: about 22.8 ktHM

North and South America: about 72 ktHM

Asia and Africa: about 17.7 ktHM

The total amount is, as mentioned before, about 145,000 tHM.

The best estimate for the coming decades is:

Western Europe: quasi-constant quantities (even slight decrease) of spent fuel to be stored. This means less fuel discharged than reprocessed.

Asia, Africa, and Eastern Europe will quasi-double their spent fuel volume to be stored in the coming ten years.

In the Americas: quasi all fuel discharged will be stored.

2.4 Spent-fuel storage according to storage type

As you can see from Table I, various types of wet- and dry-storage facilities are operating in our member states. The current global world storage capacity is slightly more than 239,000 tHM, and thus exceeds, by about 94,000 metric tons, the capacity needed by the end of 2000.

Countries operating nuclear power plants at present are or were increasing their existing at-the-reactor storage capacity by reracking the storage pools with high-density racks, by implementing burnup credit or by commissioning AFR-storage facilities.

Globally, all types of storage facilities have excess capacity available. Figure 3 compares the capacities of the various storage types with their current inventories.

The storage capacity of new facilities, presently under construction in the various regions can be seen in Table II. The capacities under construction are limited (from about 1.5 ktHM in Asia to 6.8 ktHM in the Americas). The total present capacity growth, some 12.9 ktHM, assured by the facilities under construction, is in equilibrium with the upcoming amounts of spent fuel. Table II indicates that the dry-storage away-from-reactor sites are getting preference.

2.5 Balance between arising and existing storage capacity

We estimate that, on a worldwide basis, the spent fuel arising will fill the existing storage facilities and those under construction by around the year 2015 if no additional facilities are built by that time. (See Figure 4.)

However, there is no reason to believe that no new construction projects for storage will be launched. Consequently, we do not expect a storage shortage globally.

A worldwide or regional approach does not imply any problems. On a national level, a shortage may occur if no construction or expansion can be financed or licensed.

Indeed, nationally, the situation differs from country to country and sometimes even from utility to utility. In some cases, the storage pools are fully occupied by spent fuel allowing emergency core unloading only by special measures. Hence, additional storage capacity has to be installed in time to avoid this problem. In other cases, additional storage capacity has to be installed in time to replace wet-storage facilities which cannot be refurbished. In particular in some countries of Eastern Europe, plant operation might be jeopardized if additional storage capacity cannot be installed in time.

In the past, most of the countries in this region heavily relied on the Soviet Union for their spent-fuel management. Spent-fuel return agreements signed in the past with the former Soviet Union were abandoned or amended on a commercial basis. Due to economic constraints, most countries did not opt for the commercial contracts. As a result, many

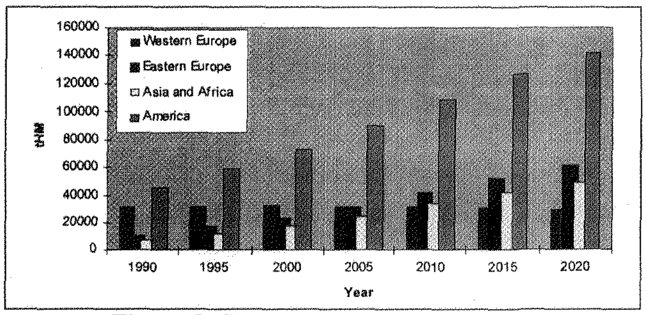

Figure 2. Spent fuel stored by regions

Table II. Spent-fuel Storage Capacity Under Construction

Region	AFR wet storage tHM	AFR dry storage tHM	Total capacity tHM
Western Europe	3,000		3,000
Eastern Europe		1,600	1,600
North America		6,800	6,800
Asia	700	800	1,500
Total	3,700	9,200	12,900

Table I. Spent-fuel Storage According to Storage Type

Region	NPP pool storage capacity tHM	AFR wet storage capacity tHM	AFR dry storage capacity tHM	Total storage capacity tHM
Western Europe	28,265	32,270	10,416	70,951
Eastern Europe	11,913	20,788	1,471	34,172
America	94,662	1,712	6,342	102,716
Asia & Africa	27,924	1,725	1,737	31,386
Total	162,764	56,495	19,966	239,225

nuclear power plants in this region are or will be faced with a shortage of spent-fuel storage capacity.

In other cases, additional storage capacity has to be installed in time to replace wet storage facilities which cannot be refurbished, as it is the case in Chernobyl.

If there will be a delay in reprocessing, more spent fuel needs to be stored. This situation would demand more storage capacity. Further postponement of the decision on the development of final repositories will also lead to a higher demand for spent-fuel storage capacity.

3. International Storage Programs

Western Europe

Belgium designed and licensed a storage capacity which corresponds to forty years operation of all NPPs. The Tihange NPP operates an AFR wet-storage facility with a pool capacity of 3,700 fuel assemblies. The dry-storage facility at DOEL can accommodate sixty casks, each cask with twenty-eight fuel assemblies. The second building is already authorized.

Finland operates AFR wet-storage facilities at each of their BWR and WWER NPPs.

France has a huge storage capacity (14,400 tHM) at the La Hague reprocessing plant used for spent fuel awaiting reprocessing. Cogema is considering extending this storage capacity to 18,000 tHM. A license application was already filed and a public inquiry was held in spring 2000. Cogema is still waiting for a license.

A dry vault storage cascade is operated at Cadarache. It is used for research reactor fuel.

Germany operates four AFR dry-storage facilities using mainly CASTOR casks and two AFR wet-storage facilities. The Ahaus dry-storage facility is licensed for 370 LWR and 305 THTR casks. Six CASTOR V LWR-casks and 305 THTR-casks are stored. The Gorleben dry-storage facility is licensed for 420 LWR and HLW casks. Five LWR casks and three HLW casks are stored. At Greifswald the dry-storage facility for WWER fuel is commissioned and licensed. It will be loaded in 2001 with the WWER fuel now stored in the Greifswald wet-storage facility. Several casks are already loaded and prepared for storage. The dry-storage facility at Juelich, licensed for 158 casks, is storing 154 CASTOR-AVR casks with the fuel from the pebble bed test reactor, which was operated at the Juelich Research Center. Obrigheim NPP commissioned an AFR wet-storage facility for 530 fuel assemblies which can be extended to 980 fuel assemblies if necessary.

Due to the new policy in Germany to avoid transportation of spent fuel, license applications were filed for thirteen AFR dry-storage facilities at the NPP sites using dual purpose casks. One license application was filed at the end of 1998, nine were filed in November and December 1999, and three in February 2000. In four cases, the license application includes also so-called interim storage. These interim storage facilities are foreseen for about twenty-four casks and are aimed to be available in a short time to bridge the time necessary to license and build the dry storage facility. The interim storage uses the same CASTOR casks that will be used in the future. The difference is the casks will be placed horizontally on a pad and covered by a concrete hood.

The NPP Emsland at Lingen, Lower Saxony, was the first to file the license application and the first to receive a license. Construction was started in October 2000. This facility will have 100 positions for storage casks. The concept is similar to the Ahaus and Gorleben storage facilities. Neckarwestheim has, due to local conditions, a different concept with two tunnels below surface.

Italy still stores 233 tHM spent fuel from the shut-down reactors. They are evaluating a dry-storage facility using storage casks.

The Netherlands will send all spent power-reactor fuel abroad for reprocessing and store the waste and research reactor fuel in a facility under construction.

Spain has reracked all reactor pools with high density racks using burnup credit. The construction of a dry cask storage facility at the Trillo site was started in early 2000 and should be commissioned by end of 2001. Trillo is the first NPP running out of storage space in 2003. The first two

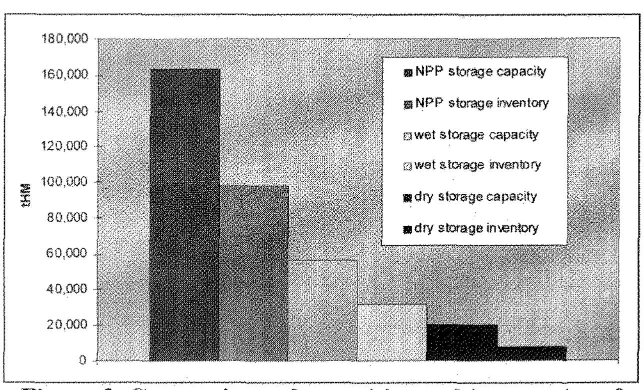

Figure 3. Comparison of capacities and inventories of different types of spent fuel storage

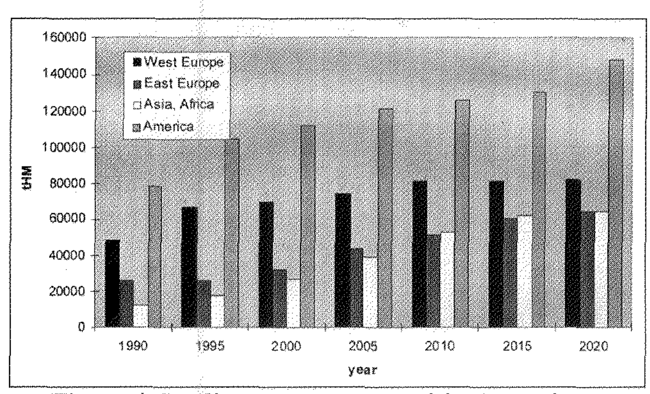

Figure 4. Predicted storage capacities by regions.

dual-purpose metallic casks were also fabricated but not yet licensed. A central dry-storage facility is planned for 2012.

Sweden operates the AFR wet-storage facility CLAB which was originally built for 3,000 tHM capacity. The use of high-density canisters brought the capacity up to 5,000 tHM. A second cavern is under construction for another 3,000 tHM, supplying sufficient capacity for the lifetime of all Swedish NPPs.

Switzerland commissioned two AFR dry-storage facilities, the central storage ZWILAG with 200 cask positions for spent fuel and HLW from reprocessing. The ZWIBEZ facility, used only for the fuel and HLW from the reactors at Beznau, has forty-eight cask positions. The first casks in Switzerland will probably be loaded in 2001.

United Kingdom uses wet storage at the BNFL reprocessing plant at Sellafield for AGR and foreign LWR fuel. Large storage pools are used to hold the spent fuel prior to reprocessing. An AFR modular vault dry storage facility is used at the Wylfa NPP with a capacity of 2x350 tHM. This facility is different from the other MVDS facilities. The fuel is stored in a conveyor. The three reactors at Wylfa have no spent-fuel storage pool. The Magnox fuel is transferred, after discharge, to dry CO_2 storage facilities with a capacity of 86 tHM each.

Central and Eastern Europe

Armenia commissioned an AFR dry-storage facility using the NUHOMS technology. Eleven NUHOMS units were built, each one capable of storing fifty-six WWER-440 fuel assemblies. Six units have already been loaded.

Bulgaria has an AFR wet-storage facility at Kozloduy. This facility was refurbished during the last few years to meet earthquake standards and has been in operation again since last year. Bulgaria planned to send back some of their fuel to Russia. Due to transit problems, this has not occurred yet. To increase storage capacity it is now planned to investigate the use of similar storage baskets as used in the Bohunice AFR wet-storage facility. This could increase the storage capacity by more than a factor of two. However, it has to be investigated if the construction of the building is able to take the higher load under consideration that earthquakes have to be taken into account. The decision to build a dry-storage facility was postponed but will be reconsidered.

Czech Republic has an AFR dry-cask storage facility using the CASTOR-WWER 440 cask with sixty cask positions equivalent to about 600 tHM. An extension of this facility by another 1,340 tHM has been decided. A new storage hall will be built parallel to the one in operation. This facility will be needed by 2006. For Temelin, a dry-storage facility with a capacity of about 1,370 tHM will be needed in the future.

Hungary has an AFR modular vault dry-storage facility. The vaults 4 to 7 went into operation in 2000. Construction of vaults 8 to 11 started in 2000. Hungary is considering changing to a different dry-storage technology in the future.

Lithuania operates at the Ignalina site an AFR dry-storage facility using the CASTOR-RBMK cask. The facility has seventy-two cask positions. In September 2000, twenty CASTOR casks (116 tHM) were loaded. Ignalina will now switch to the CONSTOR-RBMK cask which is under licensing. The cold tests with this cask have already been performed. The hot trial tests are foreseen for spring 2001 followed by regular loading. Fifteen CONSTOR casks are already available at Ignalina.

Romania plans to build an AFR dry-spent fuel storage facility with a capacity of 6000 tHM at their Cernavoda NPP site. The tendering process was initiated in late 2000 with a deadline for the offers on November 30. Five offers were presented. The time for making a decision and the construction is very short as additional storage capacity will be needed in April 2003.

Russia has operating AFR wet-storage facilities for RBMK fuel at Kursk, Leningrad, and Smolensk NPPs with a total capacity of 15,972 tHM. Other AFR wet-storage facilities for WWER fuel are operating at the Novo-Voronezh NPP with 400 tHM capacity, at the Mayak reprocessing plant with 560 tHM capacity, and at Krasnoyarsk with 6,000 tHM capacity. It is planned to use new baskets for sixteen WWER-1000 fuel assemblies at Krasnoyarsk instead of the baskets for twelve fuel assemblies. This will increase the capacity to 9,000 tHM. A reprocessing plant has been planned at Krasnoyarsk but has been postponed for the time being. The Mayak storage capacity will be extended but only for submarine fuel.

Russia is considering dry multi-vault storage for submarine and icebreaker fuel in the Arctic area and with the similar technique a central storage facility at Krasnoyarsk for RBMK fuel. Storage casks have been developed for submarine fuel but not licensed by Gosatomnadzor. The same cask type was foreseen for RBMK fuel. Due to the large number of RBMK fuel assemblies it is not likely that only casks will be used. The number of casks would be too high and so this storage technology too expensive for Russia.

Slovak Republic operates an AFR wet-storage facility at Bohunice. The facility has been renovated to meet enhanced safety criteria. As a part of the renovation the capacity was increased from 600 tHM to 1,693 tHM (14,112 fuel assemblies) by using new high-density baskets with a different shape. Mochovce can store its fuel in the at-reactor pools until 2006 in unit 1 and until 2008 in unit 2. The Slovak Republic is considering send some fuel to the Bohunice AFR storage facility. In the long run, Mochovce needs an AFR storage facility of its own.

Slovenia has a pool capacity for 470 fuel assemblies equivalent to 193 tHM at Krsko NPP. In spring 2000, the Krsko operator signed a contract for reracking the spent-fuel pool that should be completed by 2003. The installed capacity (1,750 locations) will be sufficient for plant life. In case of plant life extension, it may be possible to extend the capacity to 2,400 locations.

Ukraine currently has one AFR wet-storage facility for the RBMK fuel in Chernobyl. An AFR VSC dry-storage facility was commissioned at Zaporozhe NPP but a license has not yet been granted due to concerns that still have to be resolved. The initial capacity of this facility is fourteen VSC-24 units, 140 tHM.

Commissioning of 256 NUHOMS modules is scheduled for late 2001 at the Chernobyl site. The NUHOMS modules will be built in two parallel lines of 128 modules. This capacity is sufficient for the 25,000 RBMK fuel assemblies (2,867 tHM) of the Chernobyl site. Each module contains about 11.2 tHM. Plans are to load about 2,500 fuel assemblies per year into the storage facility.

Asia and Africa

China is commissioning a wet-storage facility with a capacity of 550 tHM at its reprocessing plant. This facility should be operational soon.

India has a wet-storage facility at Tarapur for 2,000 BWR assemblies and a dry-storage facility that uses casks. The annual spent fuel arising is about 230 tHM. However, the information currently available is limited.

Japan operates a wet and a dry AFR storage facility at Fukushima NPP. A second wet AFR storage facility is operated at the Tokai reprocessing plant. A third wet AFR storage facility with 1,600 tHM capacity has been commissioned and certified for operation at the second reprocessing plant at Rokkasho. However, this facility has not been licensed by the local authorities. There are plans to build an AFR dry-cask storage facility for long-term storage until 2010.

Kazakhstan packaged the spent-fuel assemblies from the B-350 core in special dry containers which are deposited in a salt mine for interim storage. This is a kind of dry storage, because the containers are welded and filled with Argon.

Republic of Korea has two dry AFR storage facilities in operation at Wolsong with a site capacity of 3,078 tHM. The at reactor wet storage capacity at the Kori, Yongwang, and Ulcin NPPs were increased by using high-density spent-fuel racks. Korea is considering a central wet-storage facility to be operational in 2006.

Pakistan has one reactor in operation. There is no information on their storage capacities.

South Africa plans to rerack the spent-fuel storage pool with high-density racks. ESCOM owns four CASTOR X/28/F dual-purpose casks with a dry-storage capacity of about 52 tHM which were loaded to give space in the pools for reracking.

The Americas

Argentina has a wet AFR storage facility for spent fuel at the Atucha NPP and a dry AFR storage facility for spent fuel at the Embalse NPP. At the Atucha I pool 2, a project of reracking is underway to enlarge the capacity.

Brazil currently stores its spent fuel in the reactor pool at the Angra 1 NPP.

Canada has dry AFR storage facilities at the decommissioned stations Chalk River, Gentilly 1, Whiteshell, and Douglas Point with 465 tHM in storage and at the stations in operation at Point Lepreu with concrete silos and a capacity of 722 tHM, Gentilly 2 with CANSTOR and 953 tHM capacity, and Pickering with dry-storage containers and 693 tHM capacity. Bruce will adapt a DSC-storage facility which should be operational in 2002. Additional facilities are under licensing at Pickering (stage II), and Bruce. Another facility is planned at Darlington.

Mexico stores its spent fuel in the reactor pools of the two Laguna Verde units.

United States stores most of its spent fuel in the at reactors' fuel pools. Many pools have been reracked over the years with high-density racks. Some utilities are using additional dry-storage systems at their reactor sites. Currently, more than 2,000 tHM of the spent fuel are stored under dry condition at seventeen sites in thirteen states. Several different systems are in use: NUHOMS, VSC, CASTOR-casks, TN-casks, NAC-casks, MC, and TranStor. The only AFR wet-storage facility is operated at Morris. There are another eighteen potential sites for AFR dry-storage under investigation.

4. Conclusions

I believe that, from an international organization's perspective, we can draw the following conclusions:

We have a relatively good view of the spent fuel arising from commercial reactors over the next fifteen to twenty years.

At present, there is sufficient spent-fuel storage capacity on a worldwide basis. However, nationally or on a specific site basis, the situation can be different and might need urgent attention.

The first geological repositories for the final disposal of spent fuel are not expected to be in operation before 2010. Many countries have not yet started specific site investigations. As a consequence, the use of interim storage will be the primary spent-fuel management solution for the next decades in many member states.

More spent-fuel storage capacity is required beyond 2015, if countries defer their decision to choose geological repositories.

The storage duration becomes longer than earlier anticipated, because of the selection of the wait-and-see policy chosen by many nuclear power countries. The use of higher enriched fuel with higher burnup results in higher decay heat and longer storage periods.

Experience in spent-fuel storage during about thirty years did not show any significant problems. However, much longer storage periods have to be envisaged and the additional challenges with regard to fuel and structure behavior have to be taken into account.

The Membership Committee of the INMM welcomes your contributions to the Member News section of the *JNMM*. Please keep us up to date on promotions, awards, retirements, and other career news.

Send your news and photos to Managing Editor Patricia Sullivan at INMM Headquarters, 60 Revere Drive,Suite 500, Northbrook, IL 60062, or by E-mail at psullivan@inmm.org or by fax to 847/480-9282. Be sure to include a day time phone number and e-mail address.

- Managing Director Hiroyoshi Kurihara of the Nuclear Material Control Center and Assistant Professor Yoshihiro Nakagome of Kyoto University were recognized for their outstanding contributions to nuclear material management October 26, 2000, at an event in commemoration of Japan's nuclear energy. The Japanese Minister of Science and Technology Agency presented the award.
- Dr. Joseph P. Indusi was selected by Brookhaven National Laboratory to head its Nonproliferation and National Security Department. The appointment became effective January 1, 2001. The Nonproliferation and National Security Department includes the Division of U.S.-Russian Security Program Division, the Safeguards and Arms Control Division, and the International Safeguards Project Office. Indusi holds a Ph.D. from the State University of New York at Stony Brook, and has been involved in safeguards and nonproliferation programs at BNL since his arrival in 1973. He is nationally recognized for his technical contributions to U.S. safeguards and nonproliferation programs. He is the former head of the Safeguards and Arms Control Division.

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Paducah Radiation Study Released by DOE

From 1952-1991, an estimated 2,500 to 4,000 people worked in areas of the Paducah Gaseous Diffusion Plant in Paducah, Kentucky, that increased their potential radiation exposure beyond that expected for workers elsewhere in the plant, according to a study released in January by the U.S. Department of Energy.

These highest-risk areas included the Feed Plant, the Decontamination Building, the Metals Building, and the Cascade Building. The tasks which had the most potential for increased exposure included ash handling, cylinder heels cleaning, derbies processing, pulverization operation, flange grinding, and baghouse filter changing.

The study helps to focus future health studies by identifying the job types, locations, and time period that could have posed the highest risk. The study did not attempt to estimate doses for individual workers.

While all types of possible radiation exposures were considered in the study, particular attention was given to potential exposures to transuranic elements. Current practices at the Paducah Gaseous Diffusion Plant keep worker exposures well below historic levels.

The study results are available at http://tis.eh.doe.gov.

Fellowship Honors Nuclear Pioneer

A fellowship honoring nuclear pioneer Dr. Glenn Seaborg aims to provide a definitive record of the Nuclear Era. The fellowship was announced in late November during a celebration of the 50th anniversary of the Department of Energy's Savannah River Site.

The fellowship is named in honor of the Nobel-winning pioneer of the atomic age and provides an opportunity for college students to spend a year in Washington, D.C., helping the DOE write the definitive history of this era. Dr. Seaborg, who died in 1999, was a Manhattan Project pioneer, Nobel Laureate, head of the Atomic Energy Commission, presidential advisor, University of California-Berkeley chancellor, and a respected science educator.

The Seaborg Fellowship is open to all recent American history majors currently enrolled in a doctorate program in the United States. Fellows will receive a stipend and reimbursement for round-trip transportation between Washington, D.C., and their home or campus.

The fellowship is for one academic year. For more information, contact DOE Chief Historian Skip Gosling at skip.gosling@hq.doe.gov.

NSA Cites Los Alamos for Safety Violations

The Department of Energy's National Nuclear Security Administration has cited the University of California for violations of nuclear safety rules at the Los Alamos National Laboratory in New Mexico. The University of California operates LANL for the NNSA.

The violations are described in a Preliminary Notice of Violation which was issued January 19. The violations stem from several events, including a March 2000 operational event at one facility in which eight workers were exposed to airborne plutonium during a leak from a glovebox auxiliary system. The PNOV also cites several events at a second facility in which nuclear facilities were operated outside of the limits and controls set by facility safety documents.

The PNOV was issued by John Gordon, administrator of the National Nuclear Security Administration, upon the recommendation of the Department of Energy assistant secretary for environment, safety, and health. "Our goal is to avoid such incidents by being proactive and making safety an integral part of every operation," said Gordon.

The March 2000 plutonium release occurred during the performance of corrective maintenance on a glovebox, a sealed system under negative pressure, which allows manipulation of objects inside the box via gloves integrated into the sides of the box. The plutonium was released via a leak from a loose fitting in an auxiliary gas system. Subsequent investigation identified a number of deficiencies associated with the work, including a lack of formal direction and authorization, and failure to comply with established procedures.

No immediate adverse health consequences resulted from the exposure, and involved workers were placed on temporary work restrictions to limit additional exposure until dose estimates could be determined. Up to three workers may have received exposures that exceeded the annual regulatory limit set for this work; one worker's exposure has been estimated at more than five times the annual limit.

The PNOV also addressed several events occurring after March 2000 in which similar concerns relating to procedural adequacy and implementation were identified. None of these later events resulted in worker contamination.

The second facility is where Los Alamos workers perform experiments in nuclear criticality. The investigation determined that there were problems with work controls, and with operating within the parameters that LANL had established. While no actual adverse consequences occurred because of these deficiencies, they were collectively significant because they represented an unacceptable trend in the operation and maintenance of nuclear facilities.

LANL is exempt from civil penalty by statute and no civil penalty is being assessed in association with the current PNOV. If not exempt, a civil penalty of $605,000 would have been assessed, based on the significance of the events. This penalty amount reflects partial mitigation for corrective actions taken and self-identification of one issue.

DOE to Build Isotope Facility at Tennessee State University

In January, the U.S. Department of Energy announced a joint project with Tennessee State University that will provide U.S. researchers with a reliable supply of stable, nonradioactive isotopes.

Use of stable isotopes is on the increase in the United States. Stable isotopes are invaluable in a wide array of scientific analyses, particularly for high-accuracy mass spectrometry.

Until recently, the Department of Energy provided stable isotopes to researchers and industry at its Oak Ridge National Laboratory in Tennessee and used Manhattan Project-era machines called calutrons to manufacture them. Industrial customers are now being served by overseas suppliers, leaving the existing U.S. capacity too large and too expensive for the needs of researchers. After a search for an industrial partner to operate the calutrons proved unsuccessful, DOE decided to mothball the calutrons and design a new, more cost-effective production facility. The existing machines were too large and inefficient to produce small quantities of the large variety of isotopes needed for research.

Oak Ridge National Laboratory will apply its expertise in the production of stable isotopes to assure that the new facility is installed and operated in a manner that best suits the varied needs of the U.S. research community. The new facility could be operational in approximately two years.

Tennessee State University will provide infrastructure support for the facility and work with the Energy Department to establish new educational programs for students at the university and other institutions in the region. In addition to providing its technical expertise to support the operation of the new facility, Oak Ridge National Laboratory will work closely with Tennessee State to develop a joint educational agenda in stable isotopes research.

DOE Cites Argonne National Laboratory-West for Violations

The DOE's Office of Environment, Safety and Health has cited the University of Chicago, operator of the Argonne National Laboratory-West Laboratory in Idaho Falls, Idaho, for several nuclear safety violations that took place in 2000. The enforcement action would have been accompanied by a civil penalty of $110,000, but the University of Chicago is one of the not-for-profit institutions currently exempt by statute from paying civil penalties.

One issue addressed in the Preliminary Notice of Violation concerns an April 2000 event in which a worker was contaminated while patching penetration holes at the Fuel Conditioning Facility, a facility used to prepare spent fuel for disposal. DOE found that the laboratory failed to effectively analyze the planned work activity and its associated hazards, and did not use design and administrative controls that would minimize worker exposures to hazardous materials. Although the worker intake resulting from the event was low, the activity involved the potential for significant levels of contamination.

The laboratory was also cited for failing to follow procedures for moving containers into a radiologically controlled area at the Fuel Conditioning Facility. Subsequent investigations by Argonne staff revealed that similar violations had been occurring during the past several years. While fissile material was not involved in the transfer, DOE is concerned with the long-standing and widespread nature of the violations in a radiological area, and the failure to maintain strict compliance with nuclear safety requirements.

The notice also cites the Laboratory for its failure to effectively implement a formal quality improvement effort—meaning that processes for detecting problems were not effective, root cause analyses were not routinely performed, and corrective actions were often inadequate. Problems in the quality improvement area had been repeatedly identified to Argonne management in prior DOE reviews.

The Argonne-West Notice identifies two Severity Level II and two Severity Level III violations. Level I violations represent the most significant, with actual or potential significant consequences to the worker or public. In response to the Notice of Violation (whether or not there is a civil penalty), contractors are required to document specific actions taken and planned to prevent recurrence of similar events. The Chicago Operations Office will verify completion of corrective actions before the case is closed.

DOE noted that management changes have been made since the violations took place to put greater emphasis and visibility on nuclear safety requirements. These improvements include training and strengthening procedural requirements for nuclear work. Continued management attention to the weaknesses reflected in this enforcement action will be critical in achieving substantial improvements in the safety culture for operations at Argonne National Lab-West.

Argonne National Laboratory-West is part of the Argonne National Laboratory, which is operated by the University of Chicago for the DOE.

Calendar

May 14-16
Radiation Dose Rate Management in the Nuclear Industry International Conference, Windermere, Cumbria, U.K. Sponsor: British Nuclear Energy Society. Contact: Sue Frye, Conferences Services, British Nuclear Energy Society, 1 Great George St., London SW1P 3AA; phone, 44 (0) 20 7665 2315; fax, 44 (0) 20 7233 1473; E-mail, sue.frye@ice.org.uk.

May 15-17
Annual Meeting on Nuclear Technology 2001, Kulturpalast, Dresden, Germany. Sponsors: German Nuclear Society and German Atomic Forum. Contact: Congress Office, INFORUM GmbH, Tulpenfeld 19, D53113 Bonn, Germany; phone, 49 (0)228-507 223; fax: 49 (0) 228-507 262; E-mail, tagungen@inforum-GmbH.de.

May 21-23
Nuclear Energy: Building the Future, Washington Monarch Hotel, Washington, D.C. Sponsor: Nuclear Energy Institute. Contact: Alexandra Iwuchukwu, Nuclear Energy Institute, Department 9013, Washington, D.C. 20061-9013; phone, 202/739-8039; fax, 202/872-0560; E-mail, registrar@nei.org.

May 29-31
2001 Power-Gen Europe, Brussels Exhibition Centre, Brussels, Belgium. Contact: Power Gen Europe Pennwell, Pennwell House, Horseshoe Hill, Upshire, Essex EN9 3SR, UK; phone, 44 (0)1992-656 631; fax, 44 (0) 1992-656 704; E-mail, attendingpge@pennwell.com.

June 25-28
National Space & Missile Materials Symposium, Monterey, California. Sponsor: Air Force Research Laboratory. Contact: Pat Sisson; phone, 973/254-7950; E-mail, psisson@anteon.com; Web site, http://www.usasymposium.com.

July 15-19
42nd INMM Annual Meeting, Renaissance Esmeralda Resort, Indian Wells, California, U.S.A. Sponsor: Institute of Nuclear Materials Management. Contact: INMM; phone, 847/480-9573; fax, 847/480-9282; E-mail, inmm@inmm.org; Web site, http://www.inmm.org.

September 3-7
PATRAM 2001, Chicago, Ill., U.S.A. Sponsors: U.S. Department of Energy, in cooperation with the International Atomic Energy Agency. Hosted by the Institute for Nuclear Materials Management. Chicago Hilton and Towers. Contact: INMM, phone, 847/480-6342; Web site, http://www.patram.org.

September 9-13
International Meeting on the Back End of the Fuel Cycle: From Research to Solutions (GLOBAL 2001), Paris, France. Sponsor: American Nuclear Society. Contact: American Nuclear Society Meetings Department, 555 North Kensington Avenue, LaGrange Park, IL 60526, U.S.A.; phone, 708/352-6611; fax,708/352-6464; E-mail, meetings@ans.org; Web site, http://www.ans.org/meetings.

September 17-21
45th General Conference of the International Atomic Energy Agency, Vienna, Austria. Sponsor: International Atomic Energy Agency. Contact: Conference Service Section, IAEA, P.O. Box 100, A-1400 Vienna, Austria; phone, 43 1 2600 21310; fax, 43 1 26007; E-mail, Official.Mail@iaea.org; Web site, http://www.iaea.org/worldatom/.

September 30-October 3
NRI International Uranium Fuel Seminar, South Seas Plantation, Captiva Island, Florida. Sponsor: Nuclear Energy Institute. Contact: Nuclear Energy Institute, 1776 I Street, NW, Suite 400, Washington, D.C. 20006-3708.

October 17-18
Nuclear Decommissioning (DECOM 2001) International Conference, London, England. Organized by British Nuclear Energy Society/ImechE. Contact: Maureen Carter, conference office, Institution of Mechanical Engineers, 1 Birdcage Walk, London, SW1P 3JJ; phone, 44 (0) 20 7222 7899; fax, 44 (0) 20 7222 4557; E-mail, m_carter@imeche.org.uk; Web site, http://www.imeche.org.uk.

October 29-November 1
Symposium on International Safeguards: Verification and Nuclear Material Security, Vienna, Austria. Sponsor: International Atomic Energy Agency in cooperation with ESARDA and INMM. Contact: Regina Perricos, Conference Service Section, Division of Conference and Document Services, IAEA; phone, 42 1 26000, Ext. 21315 or 21311; E-mail, R.Perricos@iaea.org; Web site, http://www.iaea.org/ world-atom/Meetings/Planned/2001.