the Archivist on recent developments at NARA.

The meeting will be open to the public. For further information, contact David Peterson at 301–713–6050.

Dated: August 31, 2001.

Mary Ann Hadyka,

Committee Management Officer.

[FR Doc. 01–22483 Filed 9–6–01; 8:45 am]

BILLING CODE 7515–01–U

NATIONAL SCIENCE FOUNDATION

Advisory Committee for Social, Behavioral, and Economic Sciences; Notice of Meeting

In accordance with the Federal Advisory Committee Act (Pub. L. 92–463, as amended), the National Science Foundation announces the following meeting:

Name: Advisory Committee for Social, Behavioral, and Economic Sciences (1171), NSF.

Date/Time: September 21, 2001; 8:30 a.m.–5 p.m.

Place: National Science Foundation, Room 970, 4201 Wilson Blvd., Arlington, VA.

Type of Meeting: Open (Members of the public who wish to attend should arrange access ahead of time with the contact person listed below).

Contact Person: Dr. Stuart Plattner, Program Director; Division of Behavioral and Cognitive Sciences, NSF, Suite 995; 4201 Wilson Blvd., Arlington, VA 22230. Telephone: (703) 292–8740.

Minutes: May be obtained from the contact person listed above.

Purpose of Meeting: To provide advice and recommendations to the National Science Foundation on issues related to the use of human subjects in social and behavioral research.

Agenda

Discussions addressing the following topics:

Foreign Institutional Review Boards (IRBs)
Training (for principal investigators, research personnel, IRBs)
Consent (forms, signing, group/individual, students as research subjects)
Ethnography/oral history; ''ethical proofreading''
Confidentiality/privacy
Secondary subjects/secondary data; linking data
Expanding the ''exempt'' category
Deception
Subpart ''D'' of the Common Rule
Research on the World Wide Web
Data archiving

Dated: September 4, 2001.

Susanne Bolton,

Committee Management Officer.

[FR Doc. 01–22518 Filed 9–6–01; 8:45 am]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–289]

AmerGen Energy Company, LLC; Three Mile Island Nuclear Station, Unit 1; Exemption

1.0 Background

The AmerGen Energy Company, LLC (the licensee) is the holder of Facility Operating License No. DPR–50 which authorizes operation of the Three Mile Island Nuclear Station, Unit 1 (TMI–1). The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of a pressurized-water reactor located in Dauphin County in Pennsylvania.

2.0 Request/Action

Title 10 of the Code of Federal Regulations (10 CFR), part 50, Appendix G requires, in part, that pressure-temperature (P/T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR part 50, Appendix G states that ''[t]he appropriate requirements on * * * the pressure-temperature limits and minimum permissible temperature must be met for all conditions.'' Appendix G of 10 CFR part 50 specifies that these limits be at least as conservative as those obtained by following the methods of analysis and the margins of safety of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G.

Pressurized-water reactor licensees have installed cold overpressure mitigation systems/low temperature overpressure protection (LTOP) systems in order to protect the reactor coolant pressure boundary (RCPB) from being operated outside of the boundaries established by the P/T limit curves and to provide pressure relief of the RCPB during low temperature overpressurization events. The licensee is required by the TMI–1 Technical Specifications (TS) to update and submit the changes to its LTOP setpoints whenever the licensee is requesting approval for amendments to the P/T limit curves in the TMI–1 TS.

By an application dated March 29, 2001, the licensee requested amendments to the P/T limit curves in the TS. In the same application, the licensee requested an exemption from application of specific requirements of 10 CFR part 50, Appendix G, and 10 CFR part 50, Section 50.61(a)(5), in order to address provisions of amendments to the TS P/T limits curves. Specifically, the exemption would instead allow the use of ASME Code Cases and an alternative approach as follows:

1. Code Case N–588, which permits the use of circumferentially-oriented flaws in circumferential welds for development of P/T limits,
2. Code Case N–640, which permits application of the lower bound static initiation fracture toughness value equation as the basis for establishing the P/T curves in lieu of using the lower bound crack arrest fracture toughness value equation, and
3. The master curve approach for determining the initial reference temperature value for weld metal WF–70 in the TMI–1 reactor vessel.

3.0 Discussion

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, when (1) the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) when special circumstances are present. The three exemptions and their associated special circumstances are discussed below.

3.1 Code Case N–588

The licensee has proposed an exemption to allow use of ASME Code Case N–588 in conjunction with ASME Section XI and 10 CFR Part 50, Appendix G, to determine P/T limits for TMI–1. The proposed amendment to revise the P/T limits for TMI–1 relies in part on the requested exemption. These revised P/T limits have been developed using postulated flaws in the circumferential orientation for the circumferential weld in the TMI–1 RPV, in lieu of postulating axial flaws in the circumferential welds.

The use of circumferential flaws in circumferential welds is more appropriate than the use of axial flaws in circumferential welds. Since the flaws postulated in the development of P/T limits have a through-wall depth of one-quarter of the vessel wall thickness (1.94 in. for the TMI–1 RPV), the length of the postulated flaw, six times the depth, is more than 11 inches. For the circumferential weld in the TMI–1 RPV, an axial flaw of this length centered at the weld would place the tips of the postulated flaw within the adjacent base metal above and below the weld.

Therefore, the only way to maintain a flaw within the circumferential weld metal is to postulate a circumferential flaw within the weld, as accomplished using Code Case N–588. For the base metals adjacent to the circumferential welds, axial flaws are and continue to be postulated for the development of P/T limits.

The underlying purpose of ASME Section XI and 10 CFR Part 50, Appendix G, is to ensure that (1) the RCPB be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized and (2) P/T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N–588 to determine P/T operating and test curve limits per ASME Section XI, Appendix G, provides appropriate, conservative procedures to determine limiting maximum postulated defects and to consider those defects in the P/T limits. This application of the code case maintains the margin of safety for circumferential welds equivalent to that originally contemplated for plates/forgings and axial welds. Therefore, pursuant to 10 CFR 50.12(a)(2)(ii), application of the code case would continue to achieve the underlying purpose of the rule, and application of 10 CFR part 50, Appendix G in these circumstances is not necessary to achieve that purpose.

3.2 Code Case N–640

The licensee has proposed an exemption to allow use of the ASME Code Case N–640 in conjunction with ASME Section XI and 10 CFR part 50, Appendix G, to determine P/T limits for TMI–1. The proposed license amendment to revise the TS P/T operating limits for TMI–1 relies, in part, on the requested exemption. These revised P/T operating limits have been developed using the K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A–2200–1, in lieu of the K_{IA} fracture toughness curve of ASME Section XI, Appendix G, Figure G–2210–1, as the lower bound for fracture toughness. The other margins involved with the ASME Section XI, Appendix G process of determining P/T limit curves remain unchanged.

Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of the P/T operating limits curve is more technically correct than using the K_{IA} curve. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The licensee has determined that the use of the initial conservatism of the K_{IA} curve when the curve was codified in 1974 was justified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{IA} curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P/T curves based on the K_{IC} curve will enhance overall plant safety by opening the P/T operating window with the greatest safety benefit in the region of low temperature operations. The operating window through which the operator heats up and cools down the reactor coolant system (RCS) is determined by the difference between the maximum allowable pressure determined by Appendix G of ASME Section XI, and the minimum required pressure for the reactor coolant pump (RCP) seals adjusted for instrument uncertainties.

Since the RCS P/T operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME Section XI, Appendix G procedure, continued operation of TMI–1 with these P/T curves without the relief provided by ASME Code Case N–640 may unnecessarily restrict the P/T operating window, especially at low temperature conditions. The operating window becomes more restrictive with continued reactor vessel service. Implementation of the proposed P–T curves, as allowed by ASME Code Case N–640, does not significantly reduce the margin of safety. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served, and application of 10 CFR Part 50, Appendix G, in these circumstances is not necessary to achieve that purpose.

In summary, the ASME Section XI, Appendix G procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements by application of ASME Code Case N–640, while maintaining, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

3.3 Master Curve Approach

The licensee has proposed an exemption from 10 CFR Part 50.61(a)(5) to allow the use of the master curve approach as an alternative to Paragraph NB–2331 of the ASME Code to determine the initial reference temperature (RT_{NDT}) value for weld metal WF–70 in the TMI–1 reactor vessel. The evaluation was part of a pressurized thermal shock (PTS) reevaluation for the TMI–1 RPV.

The current Charpy V-notch and drop weight-based methodology described in NB–2331 establishes an RT_{NDT} value and then relies on surveillance data from the testing of Charpy specimens and/or general material embrittlement models incorporated into Regulatory Guide 1.99, Revision 2 to predict the amount this value will shift due to a given level of neutron radiation exposure. This ''initial plus shift'' methodology has been consistently used to assess RPV embrittlement in the U.S. The master curve approach, however, proposes that ''direct measurement'' of fracture toughness can be made on unirradiated specimens.

The unirradiated RT_{NDT} for WF–70 weld metal was determined from drop weight tests and fracture toughness tests from welds fabricated with WF–70 and WF–209–1 weld metal. Since WF–70 and WF–209–1 welds were fabricated using the same heat number of weld wire and the same type of flux, their material properties are considered equivalent. Charpy V-notch impact and drop weight tests (the current methodology) were applied to the WF–70 weld metal by the licensees for Zion Nuclear Power Station, Units 1 and 2, and Oconee Nuclear Station, Units 1, 2, and 3, in the early 1990s for a PTS evaluation. The tests resulted in wide variability in RT_{NDT} values. The staff concluded that the large uncertainty in RT_{NDT} values for WF–70 weld metal is due to the low upper-shelf behavior of the material. Therefore, the definition of RT_{NDT} in the ASME Code is not applicable for WF–70 weld metal due to the large variability in RT_{NDT} values. In lieu of using Charpy V-notch and drop weight data, the licensee proposed to determine the initial reference temperature value using the test results from the master curve methodology. Since the licensee did not follow the method in Section III of the ASME Code, the methodology for determining the RT_{NDT} of WF–70 does not meet the requirements of 10 CFR 50.61 and requires an exemption.

By letter dated February 22, 1994, the NRC approved the use of the master curve approach for the Zion Nuclear Power Station, Units 1 and 2, and the RT_{NDT} value is −26 °F for WF–70 weld metal. The exemption approval for the Zion station also stated that other procedures for determination of RT_{NDT} may serve as acceptable alternatives to NB–2331 contingent on staff review and approval. The staff acceptance of the alternative procedure in that evaluation was based, in part, on the analysis of a significant amount of fracture toughness data for the WF–70 weld metal. Therefore, since TMI–1 used the same weld metal as Zion and the data considered for the Zion exemption resulted in a more representative RT_{NDT} value, the TMI–1 use of the master curve approach for WF–70 weld metal is acceptable.

In summary, the underlying purpose of 10 CFR 50.61 is to ensure that the RPV is adequately protected from PTS. Application of the master curve approach to determine the unirradiated RT_{NDT} value for weld metal WF–70 is acceptable because the master curve approach is more appropriate for material with low upper-shelf behavior like WF–70 weld metal.

Therefore, pursuant to 10 CFR 50.12(a)(2)(ii), application of the master curve approach to determine the unirradiated RT_{NDT} value for weld metal WF–70 would continue to achieve the underlying purpose of the rule, and application of the definition of $RT_{NDT(U)}$ in 10 CFR 50.61(a)(5) in these circumstances is not necessary to achieve that purpose.

4.0 Conclusion

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemptions are authorized by law, will not endanger life or property or common defense and security, and are, otherwise, in the public interest. Also, special circumstances are present. Therefore, the Commission hereby grants AmerGen Energy Company, LLC exemptions from the requirements of 10 CFR part 50, Appendix G, and 10 CFR part 50, § 50.61(a)(5), for TMI–1.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (66 FR 45874).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 30th day of August 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–22514 Filed 9–6–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 72–8]

Calvert Cliffs Nuclear Power Plant; Notice of Docketing of the Materials License SNM–2505; Amendment Application for the Calvert Cliffs Independent Spent Fuel Storage Installation

By letter dated July 26, 2001, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), submitted an application to the Nuclear Regulatory Commission (NRC or the Commission) in accordance with 10 CFR part 72 requesting an amendment of the Calvert Cliffs independent spent fuel storage installation (ISFSI) license (SNM–2505) for the ISFSI located in Calvert County, Maryland. CCNPP is requesting Commission approval to amend SNM–2505 to reflect revised fuel assembly integrity analysis as described in the Safety Analysis Report. CCNPP proposed changes to Technical Specification 2.3 to remove the 15-inch drop height limit and require inspection after any drop of a dry shielded canister. CCNPP also proposed a change to Technical Specification 6.3 to revise the reference to a semi-annual environmental reporting period to be consistent with the annual reporting requirements of 10 CFR 50.36a(2).

This application was docketed under 10 CFR part 72; the ISFSI Docket No. is 72–8 and will remain the same for this action. The amendment of an ISFSI license is subject to the Commission's approval.

The Commission may issue either a notice of hearing or a notice of proposed action and opportunity for hearing in accordance with 10 CFR 72.46(b)(1) or, if a determination is made that the amendment does not present a genuine issue as to whether public health and safety will be significantly affected, take immediate action on the amendment in accordance with 10 CFR 72.46(b)(2) and provide notice of the action taken and an opportunity for interested persons to request a hearing on whether the action should be rescinded or modified.

The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These documents may be accessed through NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/nrc/adams/index.html.* If you do not have access to ADAMS or if there are problems in accessing documents located in ADAMS, contact the NRC Public Document Room Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 29th day of August 2001.

For the Nuclear Regulatory Commission.

E. William Brach,

Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 01–22515 Filed 9–6–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–334 and 50–412]

FirstEnergy Nuclear Operating Company, et al., Beaver Valley Power Station, Unit Nos. 1 and 2; Notice of Issuance of Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 241 and 121 to Facility Operating License Nos. DPR–66 and NPF–73, respectively, issued to FirstEnergy Nuclear Operating Company, et al. (the licensee), which revised the Technical Specifications (TSs) and authorized revisions to the Updated Final Safety Analysis Report (UFSAR) for operation of Beaver Valley Power Station, Unit Nos. 1 and 2, located in Shippingport, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment authorized revisions to the BVPS–1 and 2 UFSAR design-basis fuel handling accident (FHA) dose consequence analyses. The amendment also revised the BVPS–1 and 2 TSs associated with the requirements for handling irradiated fuel assemblies in the reactor containment and fuel building and the TS requirements associated with ensuring that UFSAR safety analysis assumptions are met for a postulated FHA. The term ''recently irradiated'' fuel is defined in the applicable TS Bases as ''fuel that has occupied part of a critical reactor core within the previous 100 hours.'' The purpose of the addition of the term ''recently irradiated'' throughout the TSs is to establish a point where operability of those systems typically used to mitigate the consequences of an FHA is no longer required to meet the