UNIVERSITÀ DI PISA

SICUREZZA NUCLEARE

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CdS	INGEGNERIA NUCLEARE
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CFU	12

Moduli	Settore/i	Tipo	Ore	Docente/i
ANALISI DEGLI INCIDENTI NEGLI IMPIANTI NUCLEARI	ING-IND/19	LEZIONI	60	WALTER AMBROSINI ANTONIO MANFREDINI
SICUREZZA DEGLI IMPIANTI NUCLEARI	ING-IND/19	LEZIONI	60	WALTER AMBROSINI MARCO NICOLA CARCASSI

Obiettivi di apprendimento

Conoscenze

- Conoscenza di tecniche di PRA / PSA
- Conoscenza di metodologie e principi di base della sicurezza nucleare
- Conoscenza di fenomeni osservati o che si suppone possano accadere negli impianti nucleari durante condizioni normali e accidentali, inclusi gli incidenti gravi (severi)
- Conoscenza delle salvaguardie ingegneristiche e delle strategie di mitigazione degli incidenti severi
- Conoscenza dei principi di funzionamento degli strumenti atti ad analizzare gli incidenti negli impianti nucleari
- Conoscenza dei dettagli degli eventi incidentali che si sono presentati negli impianti nucleari di potenza e degli incidenti di criticità
- Conoscenza dei principi di sicurezza e della cultura della sicurezza nucleare

Una lista più dettagliata di conoscenze è fornita a livello di illustrazione del programma.

Modalità di verifica delle conoscenze

Esame Orale con domande aperte, della durata di circa un'ora. In alcuni casi, se questo facilita lo studente, può essere assegnato un primo elenco di domande scritte in relazione alle quali lo studente può prendere appunti per poi discutere oralmente i relativi concetti.

Capacità

Il corso fornisce le seguenti abilità (capacità):

- valutazione dell'affidabilità e della disponibilità di sistemi diversi, con e senza riparazione;
- capacità di identificare gli incidenti che possono avvenire in vari tipi di impianti nucleari e di valutarne quantitativamente la loro severità;
- capacità di stimare quantitativamente fenomeni di base, come la pressurizzazione del sistema di contenimento nei reattori PWR e BWR dopo un incidente di perdita di refrigerante (LOCA), il rilascio di gas radioattivi nel gap del combustibile, bilanci energetici di base nel sistema di contenimento;
- comprensione delle tenciche utilizzate da codici di sistema e a parametri concentrati per l'analisi del comportamento del primario e del contenimento di impianti nucleari;
- capacità di discutere i principi di base per la prevenzione degli incidenti negli impianti nucleari di potenza e degli incidenti di criticità.

Modalità di verifica delle capacità

Esame orale, con assegnazione di semplici problemi

Comportamenti

Il corso ha grande rilevanza per la formazione di ingegneri nucleari con una tipica mentalità in favore della sicurezza; in questo senso è un corso che tende ad ispirare quel cambiamento di atteggiamento personale che è necessario per la "nuclear safety culture" (cultura della sicurezza nucleare). Si favoriscono atteggiamenti come il senso di responsabilità personale, apertura e comunicazione, trasparenza, mettere in luce problematiche di sicurezza tramite continue domande sulla correttezza delle pratiche e delle procedure. Alcuni elementi di base della "cultura della sicurezza nucleare" (si veda il concetto come introdotto dalla IAEA) sono inclusi specificamente in alcune lezioni, ma vengono ripetuti in ogni occasione utile durante l'intero corso.

Modalità di verifica dei comportamenti

L'esame orale ha lo scopo di evidenziare l'aatteggiamento personale proponendo domande e problemi legati alla cultura della sicurezza nucleare.

Prerequisiti (conoscenze iniziali)

Conoscenze di base relative agli impianti nucleari, alla termoidraulica, alla fisica dei reattori nucleari, alla cinetica neutronica, ai materiali nucleari.

Indicazioni metodologiche

Gli studenti sono invitati a studiare con devozione questa materia, come una delle più interdisciplinari tra i corsi proposti, che è anche un punto di arrivo per molte delle materie studiate in precedenza. La sicurezza è un aspetto della più grande importanza nella tecnologia nucleare ed il corso è concepito per fornire le conoscenze, le abilità e gli atteggiamenti necessari per sviluppare un approccio responsabile nel considerare gli impianti nucleari di potenza, le loro potenzialità e i loro rischi.

Programma (contenuti dell'insegnamento)

Modulo di Sicurezza degli Impianti Nucleari_

(si riporta il testo in lingua inglese per una migliore corrispondenza con il registro delle lezioni)
Concept of safety and difference between safety, security and safeguards: excerpts from the IAEA nuclear safety glossary. General Nuclear safety objective, Radiation protection objective and technical safety objective. Safety of operators, population and economic safety.
Nuclear safety since the Fermi Pile: diversity in SCRAM, safety culture. Milestones in nuclear safety: from the Energy Act in USA to Fukushima. The "defence in depth" and the "multiple barrier" concepts. Accidents and incidents from the IAEA safety glossary; first discussion of the INES scale.
The decay heat power as one of the most important safety concerns: typical values. Risk as a measure of safety. Definition of risk as probability of adverse events by the magintude of consequences per event. Tools for safety analysis: overview of fault trees and event trees and their use. First suggestions about common mode failures. Mention of the main reporrts concerning safety: WASH-1400, German Risk Study, NUREG-1150. USA Safety goals and first discussion about the "acceptable" risk. A concluding metaphora about safety culture: definitions of safety culture by IAEA and INPO; "traits" of a healthy nuclear safety culture by INPO: individual commitment to safety, management commitment to safety, Management systems. Nuclear Safety cilture as the goal of this course.
USA Nuclear Safety Methodology and Legislation. NRC and its roles. Safery Standards, Safety Guides, Safety Practices, Safety Reports, etc. Role of the regulations, of regulatory guides and of technical codes. 10 CFR 20, 50 and 100 in general terms. 10 CFR 50 and the issues it addresses. PSAR, FSAR, ER, SRP (classical documents for licensing). Appendix A (General Design Criteria), Appendix B (Quality Assurance), Appendix K (ECCS modelling and criteria).
Detailed comment of the Appendix A, Appendix B and Appendix K of 10 CFR 50. It is recommended to the students to have a personal reading of the documents that can be found from the websites and links introduced in the slides. The reading of General Design Criteria should be connected to the students' knowledge of pricniples applied in the design of nuclear power plants as redundancy of systems, need of a containment systems, etc. (these are just examples)
Theory and reliability applications to safety analysis. Main variables: Failure rate, Reliability, Unreliability, Availability and Unavailability. Classification of components failure. Formulae for Reliability, Unreliability and Mean Time before failure (MTBF)
Introduction to Availability. Maintainability and MTTR. Formulas of Unavailability for components with revealed failures and for components that not revealed failures.
Reliability applied to the logic: Series, Parallel, Reserve and Majority. Determination of R (Reliability) and MTBF (Mean Time Between Failure), and theorem of Bayes. Summary of Availability. Maintainability and formulas of Unavailability of safety and protection systems due to detected faults and not revealed failures.
Introduction of Logic Trees: Fault Tree and Event Tree. Construction (qualitative and quantitative) and meaning of fault trees. Minimal cut set technique and Single Failure concept.
Safety Methodologies based on reliability. Farmer's methodology and its main principles. Use of the event trees to set up histograms of probability vs. releases. Risk acceptability curve and its evolution. Simple example of application ot evaluate the maximum unvailability of a system. Criticisms and observations to Farmer's methodology. Classical Canadian Methodology based on serious event occurrence and two protection systems.
The data for the calculation of the unavailability of components and initiating event of a sequence. Databases. Databases components and their use with practical examples with the bank PSAPACK date. CCF (common Causes Failures).
Canadian safety methodology and its deterministic evolution. Safety characteristics of recent CANDU reactors. Italian probabilistic methodology by Galvagni; classes of process systems and of protection systems; indices characterising each of them and risk curve obtained by adding the indices for the initiating event and of the failed protection systems. General discussion on risk perception and acceptability: the subjective perception of risk, the individual riks and the societal risk. Cognitive map of risk by Slovik. Aspects of perceived risk and risk management. Conclusions about risk assessment (cmpletion of the presentation Annex A - Risk concepts and assessment).
Nuclear Italian Legislation: an excursus on the different laws appeared in the last decades. Description of the legislative decree 230/1995 as an example of application of rules for licensing, in its different steps, emergency planning, etc. Few notes on the law n. 99 July 23, 2009, legislative decree February 10 2010, n. 31, D. Lgs. 45, March 4, 2014, D. Lgs. 137, Sept. 15, 2017. The student is invited not to remind the single provisions, most of which are recently substituted or deleted, but to understand the complexity of regulations for licensing of nuclear power plants and their decommissioning.
Sources of European nuclear legislation. Few hints about the new nuclear safety directive and the "stress tests": what where the stress tests,

which were the most important results; importance of Education and Training in the nuclear fields according to the directive.
The Rasmussen Report: the context of the maximum credible accident (WASH 740); aspects of realism and conservativism in the Rasmussen Report. Concepts of determinisitc and probabilistic safety analysis from the IAEA glossary on safety. Proposal of a reading guide of the report presented in the slides. In Chapters 1 and 2: presentation of the different phases of the study; basic oncept of risk: social risk and individual risk; people attitudes towards individual risk from 10^-3 to 10-^6 per year in general terms. High probability events and low probability events: difference between them in terms of data availability. Different categories of risks: early fatalities, early ilnesses, late effects and property damage. Comments to Chapter 3. Defence in depth and characteristics of the considered plants (Surry PWR and Peach Bottom BWR). Location and amount of radioactivity in the plant. Adopted classification of LOCAs. Safety functions: RT, ECC, PARR, PAHR, CI. Attention to the modes of containment failure. Anticipated and unanticipated reactor accidents. Events regarding the Spent Fuel Storage Pool. Chapter 4 and its content. The tasks of the report: I) probability and magnitude of releases; II) Consequences of releases; III) Risk evaluation. Interconnection between Task I and II; Role of Commom mode failures. Definition of common mode and common cause failures as per the IAEA Safety Glossary. Evaluation of the release magnitudes and treatment of 54 nuclides by the CORRAL code. Evaluations of the consequences of releases. Ways of dose assumption by people and dose-effect relationship. Release categories for PWRs (9) and BWRs (5). Main contributing sequences in terms of probability and release categories in PWRs and BWRs. Comparison of the risks from nuclear power plants with risks from other man-made and natural causes. Short description of the NUREG-1150 report and of its methodology and results. Conclusions on the innovative features of the Rasmussen Report (or Reactor Safety Study or WASH-1400) and of its importance for the development of PRA / PSA techniques and the assessment of risk from nuclear power plants in comparison with other risks to which we are exposed.
Probabilistic Safety Analysis (PSA) of nuclear power plants: PSA levels and related objectives; objectives and use of I level PSA according to IAEA guide SS 50-P-4; objectives and use of II level PSA according to IAEA guide SS 50-P-8; evaluation of uncertainty of results of I and II level PSA; examples and applications.
Introduction to the Hydrogen Risk in Nuclear Power Plants. The source term (Zr and Steel reactions with steam, radiolysis, core concrete interaction and paint corrosion). The main characteristics of the combustion of hydrogen that affect the mitigation systems. The mitigation systems of hydrogen risk in the nuclear field; deliberate ignition, pre/post inerting, recombination. Description of such systems and their use/limitations in case of accident.
Atmospheric dispersion of gaseous pollutants. Main phenomena. Parametric effects o wind speed, stability of atmospheric conditions. Detailed explanation about the adiabatic temperature gradient and its relation to the stability in "subadiabatic" and instability in "superadiabatic" temperature gradient conditions. Mixing layer and "capping" inversion. Pasquill Gifford stability classes for day-time and night-time. Eulerian and Lagrangian coordinate systems and gaussian plume dispersion model. Reflection from the ground as the effect of a virtual source. Dispersion parameters (sigma_x and sigma_y) as a funcrion of the different stability classes and the distance from the emission site. Concentration at the gound for different stability cases and release height. Effective release height and practical formulas for its evaluation. Effect of wind speed variation. Effect of buildings: "downwash" and the virtual source method for its evaluation. Fumigation: phenomenon and evaluation formulas. Mixing layer and its evaluation by an abacus by Smith and Hunt. Phenomena of dry and wet deposition: deposition velocity and scavenging coefficient. Annual Average concentration evaluated on the statistics of wind direction index, of stability classes and wind speed classes. Wind Roses. General considerations on the case of releases of gases and vapour mixtures heavier than air. Slumping phenomena and effect of the Richardson number on dispersion. Evaluation of the Radiation Doses due to Radioactive Releases: general effects to be considered and pattern of interaction of dispersion in atmosphere and in water: inhalataion, external exposure, exposure because of depisition, ingestion by foodstuff, external irradiation from sediments. Irradiation by sewage discharge.
Dose calculations from different pathways from the IAEA safety report No. 19. Doses from direct exposure to (immersion in) the cloud (semi-infinite cloud model), dose to skin, from deposited activity and sediments, inhalation and ingestion, sewage sludge. Related tables of coefficients and constants for "occupancy", "dose conversion factors", "intake rates" etc.
Illustration of the IAEA activities both in general and for what concerns the siting of the Nuclear Plants. Vision of some presentation from the IAEA site regarding this matter. In-depth illustration of the FUNDAMENTALS and INSAG 22.
Introduction, and story, of the problem of Siting and Emergency. Logical structure of technical regulations and the time duration of each particular study.Description and comment of the characteristics of the sites to be considered in accordance with the IAEA methodology. Discussion of the UNICEN methodology, with identification of exclusion criteria, to be used in the phase of the Site Survey. Other characteristics of the sites relevant even if they are not "exclusion" parameters. Identification of the relevant numerical data regarding the criteria for exclusion. Maps DISP 77 on the status of the Italian territory in relation to the Italian methodology for the Siting of the foreseen Nuclear Energy Plan at that time. Specific Requirements – External Human Induced Events and Specific Requirements – External Events - Earthquakes
The nuclear emergency planning; common aspects and differences with chemical emergencies; main factors which affect the consequences of a nuclear accident; armamentarium of the emergency measures and their applicability in various stages of development of the emergency. The main requirements from App. E to 10CFR50; emergency planning and concept of EPZ (Emergency Planning Zones); declaration and extent of the emergency; effectiveness of the various emergency measures.
ENVIRONMENTAL IMPACT OF VARIOUS ENERGY CYCLES. Introduction to the concepts of internal and external costs. Problems in evaluating the latter and "internalising" them. Various methodologies used initially by the NRC for the Environmental Impact Assessment. Externe Projects and their methodologies, results and uncertainties. Conclusion of the cost-risk-benefit analyses.
Nuclear Safety Culture (NSC). Definitions according to INSAG and INPO. INSAG-4, INSAG-15 and INPO reports on NSC. Major components of Nuclear Safety Culture according to INSAG and detailed comments to each one of them. Excerpts from INSAG-15 on key issues in NSC. Web and video resources on NSC.
Modulo di Analisi degli Incidenti negli Impianti Nucleari
(si riporta il testo in lingua inglese per una migliore corrispondenza con il registro delle lezioni)
NPP accidents and design of ECCS. Operating states for which the reactor is designed. Reference classification of accidents (in USA) for moderate frequency, infrequent and highly unlikely accidents. General cosndierations on LOCA phenomena and ECCS in 2nd generation PWRs and BWRs. Criteria for ECCS evaluation and definition of a LOCA and of an ECCS evaluation model.
Phenomena of LOCA in PWRs. LBLOCA: typical trends of pressure, flow rate, cladding temperature and water level; description of the various phenomena: subcooled blowdown and metastability; saturated blowdown; rdistribution of energy in the fuel rod after DNB; refill and reflood periods and related issues. SBLOCA: role of stratification, natural circulation, reflux condensation, etc..
Classical description of LBLOCA phenomena in BWRs (core flow rate, T_cladding, pressure trends) for a recirculation line and a steam line break. Quick reminders about basic phenomena involved in LOCA: Critical Heat Flux (pool and flow boiling, DNB and Dry-Out), bottom-up and top-down rewetting, flooding and CCFL, critical flow. Methods for the discretisation of the fuel rod and transient redistribution of heat at CHF.

General features of the Yamanouchi model for the determination of the quench front velocity (reminders of important dimensionless numbers, Re, Pe, Bi). Need for a 2D "moving mesh" discretisation in system codes for quench front propagation analysis. General description of the coverage of ECCS for different spectra of break sizes. Assumptions from Appendix K of 10 CFR 50 for evaluating the effectiveness of ECCS. First concepts of Best Estimate Plus Uncertainty Analyses (BEPU).
Again on Evaluation Models and on Best Estimate models. Need of uncertainty evaluation with BE models in order to extrapolate the "accuracy" in evaluating data from experimental facility to the full scale reactor. Short mention of existing uncertainty methodologies. ECCS in AP1000, EPR and ESBWR: passive vs. active systems and different needs in modelling, owing to the greater coupling between primary system and containment in the case of passive reactors. (WALTER AMBROSINI)
Description of the RELAP5 code as an example of thermal-hydraulic system code. Purposes of a system code and general structures. Volumes and Junctions in staggered mesh arrangement. Balance equations for mass, momentum and energy and related jump conditions across the liquid-vapour interface. Need for Equations of State (EoS) and constitutive (or closure) relationships. Relation of the closure relationships to flow regime maps.
Flow regime maps of RELAP5 for vertical and horizontal flow; special flow regime maps for high kinetic energy (pumps) and ECC mixer. Constitutive models for interfacial and wall friction and for localised pressure drops. Wall and interfacial heat transfer. Cross flow junctions and flooding / CCFL models. Gallery of component models: branches, separators, jet-mixers, pumps, turbine, valves, accumulators, ECC mixer. Few remarks on Semi-implicit and nearly-implicit numerical schemes: Courant limitation. Heat conduction in structures: radial for normal structures and radial and axial for rewetting phenomena. General conclusions on system codes.
Calculation of the radial temperature distribution inside a fuel rod with classical assumptions and the electrical analogy. Temperature differences in the fuel, in the gap, in the cladding and in the thermal boundary layer. Derivation of the longitudinal coolant distribution in a PWR core with sinusoidal power distribution and combination of the different formulations for providing axial distributions of the relevant fule and cladding temperatures. Distribution of an Excel file combining these formulations for parametrical analyses.
Transient Thermal analysis of a nuclear fuel rod with a 1D finite volume method developed by the teacher and implemented in a Matlab script. Description of the development of the algorithm and use of the programme for analysing relevant cases: reaching different steady states with various values of power and of the boundary conditions; analysis of the energy redistribution occurring after a LOCA when DNB occurs after scram; free analyses left to the students.
Severe accident phenomena in LWRs. General remarks on the motivation to study severe accidents: early considerations from WASH 1400, TMI-2 occurrence, extension of the defence in depth to severe accident management. Synoptic view of the phenomena occurring in a severe accident in LWRs and of the models necessary to predict them, as from those adopted in the ASTEC code. Types of severe accidents: Reactivity Induced Accident (RIA) and Core Uncovery Accident (CUA). High pressure and low pressure scenarios. In-vessel phenomena and temperature scale for the major metallurgical events. Different ranges of temperature for the phases of melting and relocation of control rods and burnable poisons, creation of blockages and pools, for localized fuel damage, extended core damage and total destruction of the core. Visual description of phases of candling, blockages and pool formation.
Core relocation mechanisms. Debris relocation mechanisms. Lower head failure modes. Ex-vessel phenomena: Direct Containment Heating (DCH), Molten Core-Concrete Interaction (MCCI), Ex-Vessel steam explosion, Hudrogen risk and the Shapiro diagram.
Severe accident management guidelines and techniques: list of issues and of the related pros and cons. Among the other features, in-vessel and ex-vessel retention. Simple an complex solutions for the various issues.
NPP Containment Systems. Classification of containment systems and their typical structure for PWR and BWR: full pressure dry containment, pressure suppression containments systems (Mark I, Mark II and Mark III containments), single and double wall containments, ice consender containments, containments for CANDU reactors and role of vacuum building and dousing tank with sprays. Design parameters for pressure suppression containment systems. Principles of design of containment systems concerning DBA and severe accidents. Typical containment behaviour for full pressure, pressure suppression systems and CANDU reactors. Severe accident concerns and usefulness of pressure suppression systems.
Description of a code calculation for a blowdown from a primary system into a containment environment. Description of the procedure for the hand calculation of containment pressurization in a dry containment after a LB-LOCA: assumpions and solution algortihm. Distribution of an Excel file for performing the analysis. Analysis of the Excel file for calculating the pressure peak in a dry containment and its use for converging on the final temperature and pressure. Description of the procedure for evaluating the pressure in a containment with pressure suppression system: main assumtpions and calculation procedure. Description of an excel file for the analysis of the pressure suppression case and its use. Presentation of lumped parameter codes and demonstration of the application of a blowdown to containment codes (GOTHIC and FUMO), comparing the results with hand calculations.
Engineered Safety Features in NPPs. Main functions of the containment: CI, PARR, PAHR. Containment leakage rate and related tests. Gas liquid mass transfer for PARR; resistances, partition factors, different conditions for partitioning of radioactive substances in the liquid and gas phases. Derivation of the exponential laws for scrubbing of the radiactivity by spray systems. Containment spray systems and their efficiency; analysis based on drop size, mass transfer velocity, flow and number of droplets. Effect of additives to spray water (oxidation, hydrogen production, etc.). Gas treatment systems involving filtration: HEPA and ULPA system; composition and testing of typical filtering units. Types of tests. Penetration and efficiency of filters; problems related to their mounting. Phenomena determining the efficiency of hepa filtes and typycal curve of efficiency as a function of particle diameter (MPPS). Charcoal and zeolite filters.
The Source Term issues. Definition of "in containment source term" and chronological development of the concept in USA. Different reports issued on the subject (TID-14884, NUREG-0772, NUREG-1475). Inventory of readioactive materials in the recator core: equations for the analysis of their production and list of the most important ones with decay constant and relative abundance with reference to a 3000 MWth reactor core. Fission product release to the gap: method for evaluating the fraction of release owing ro recoil, diffusion, equiaxial grain growth and columnar growth on the basis of the conductivity integrals. Typical formulations for the analysis of releases for the fuel as a function of temperature. Hand calculation procedure for estimating the fraction of releases as an example of the more complex cases adopted in codes. Sample results for examinated sequences.
Purposes of the evaluation of the source term in the NUREG-1465. BMI suite of codes. 5 phases of release and related timing for the sequences addressed for PWR and BWR selected reactors. Type of releases (noble gases, halogens, alkali metals, etc.).
Reactivity induced accidents (RIA). Types of considered accidents and representative causes. Countermeasures for designing stable cores (feedback, duplicated scram system, control rod Worth limitation). Block diagram of the dynamics of a reactor core with feedbacks and reactivity control. Point kinetics equations in density and power form with 6 delayed neutron groups. Lumped parameter model for fuel and coolant for PWRs; role of void in BWRs.

Causes of RIA. Types of reactivity induced transients: quasi-static, superdelayed and superprompt-critical transients, description and possible casuses and effects. Treatment of the power peak for superprompt-critical recativity step insertion transients, without delayed neutrons and with linear energy feedback; related mathematical developments and conclusions about the parametric effects; short considerations on ramp effects. Anticipations on a exercise to be performed the next time.
Description of the MELCOR code. Models for severe accident analysis. Examples of application to Phebus experiments and TMI-2 analysis.
Exercises on RIA by a Matlab Program set up by the teacher implementing the point kinetics equations with linear energy feedback. Simulations of results reported in Fig. 5-8 and following of the textbook by E.E. Lewis on step and ramp insertions. Anaysis of the power peak, of the accumulated energy, of the reactor period, reactivity, precursor concentration, etc.. Analysis of steps and ramps with different neutron generation time (lifetime/keff).
The International Nuclear Event Scale (INES). Description of its basic principles frome the original IAEA brochure. The seven levels of the accident classification and the three issues considered for their establishment: 1) off-site impact; 2) on-site impact; 3) impact on defence in depth. Examples of accidents classified under the different categories.
Criticality accidents: criteria for prevention: rule of double contingency. Categorisation of systems: always safe, with limited safety, Others. Main control parameters: mass, concentration, geometry; secondary parameters: moderation, reflection, use of poisons. Diagrams of safe and un safe regions as a function of concentration for the different fissile nuclides. Detailed list of occurred criticality accidents up to the accident of Tokai-Mura. Description in details of some of them: Hanford, 1962; Wood River Junction, 1964; Tokai-Mura, 1999.
General remarks on nuclear reactor accidents. Location of TMI-2, Chernobyl and Fukushima accidents in terms of previous reactor-years of operation. List of energy related accidents (from the WNA website) and comparison with the single nuclear accident with casualties (Chernobyl).
Description of some accidents: Chalk River (1952), EBR-I (1955), Windscale (1957), SL-1 (1961), Fermi Reactor, Lucens, Browns Ferry.
The Three Miles Island accident. The causes of the accident and the main chronology of the events (the "stuck-open" PORV, the disabling of ECCS, the stop of the pumps, the delayed diagnosis and pump restart, the "hydorgen bubble" issue, the pressure spike in the containment).
The damage to the core as reconstructed after the accident. The lessons learned from the accident. YouTube movie from NRC on the accident and its consequences. Description of the TMI-2 accident with the aid of MELCOR and RELAP/SCDAP results. Watching a movie describing the first hours of the accident form the technical and the societal point of view.
The Chernobyl accident. The plant and its characteristics (e.g., the positive void coefficient in some operating conditions). The sequence of events and the violation of procedures and prescriptions. The New Safe Containment and the health consequences according to WHO and UNSCEAR.
The Fukushima accident; the status of the plant before the earthquake and the tsunami; the sequence of the events at each one of the 4 affected units; the health consequences; the lesson learned; suggestions for video resources.
Description of the content of Safety Analysis Report of NPPs (according to IAEA safety guides). Example of the content of the SAR of Montalto di Castro (excerpts of the index). Description of the SAR of the Caorso nuclear power plant. Example of DBA accidents considered in the report itself.
Additional considerations on latest developments on CANDU reactor safety methodology and classes of accidents; suggestions about the IDDA code and the dynamic analysis of event trees (this was a very quick overview.
Short notes on Sodium cooled fast reactors and their safety. Safety characteristics of SFRs and general safety principles to be used in their design. Plant Protection System (PPS) and general classification of transients in "unprotected" and "protected". ATWS and UTOP, ULOF, ULOHS. Role of PRA and design characteristics of the PPS. Implementation of redundancy and diversity in PPS design. Role of natural circularion. Concerns related to Fuel Element Failure Propagation. Gallery of protected and unprotected accidents. Relevance of intrinsic feedback in unpreotected transients. Hypothetical Core Disruption Accident. Characteristics and assumptions. Bethe and Tait theory. Role of Doppler effect in decreasing the releases of energy. Sodium Fires. Containment types. Suggested readings.
AP1000 Safety Analysis. Results of PSA and and application of MELCOR to a Station Black-Out and two different DVI injection line breaks.

Bibliografia e materiale didattico

- Slides del Professor Marino Mazzini, riviste ed aggiornate dal Prof. Walter Ambrosini (21 unità di slides e trattazioni aggiuntive documentate a parte)
- Slides del Prof. Carcassi e del Prof. Manfredini
- E.E. Lewis, Nuclear Power Reactor Safety, John Wiley and Sons, 1977
- G. Petrangeli, Nuclear Safety, Elsevier 2007
- Testo sulla Sicurezza Nucleare del Prof. Mazzini (Versione 2017-2018), in formato Word
- Risorse online, documenti e report suggeriti dal docente durante le lezioni e indicati nelle slides

Indicazioni per non frequentanti

Il materiale viene aggiornato ogni anno e posto in cuna cartella dropbox condivisa. Contattare via e-mail il Prof. Ambrosini all'indirizzo walter.ambrosini@ing.unipi.it per ottenere il link.
Il docente è disponibile con continuità a ricevere gli studenti, anche utilizzando strumenti telematici, per risolvere il loro problemi di apprendimento.

Modalità d'esame

Esame orale della durata di circa un'ora con la proposta di domande aperte e, nel caso, anche di qualche esercizio.

Altri riferimenti web

- http://younuclear.ing.unipi.it/
- https://www.facebook.com/NuclearEngineeringPisa/
- https://www.linkedin.com/groups/4501364
- https://www.linkedin.com/groups/8463083

Note

Per le date degli esami si consulti questo stesso portale e si contatti COMUNQUE l'indirizzo walter.ambrosini@ing.unipi.it per definire eventuali date specifiche.

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